L-2022-083
10 CFR 50.90
10 CFR 50.91(b)(1)

June 3, 2022

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
11545 Rockville Pike One White Flint
North Rockville, MD 20852-2746

RE: St. Lucie Plant Units 1 and 2
Dockets 50-335 and 50-389
Renewed Facility Operating Licenses DPR-67 and NPF-16

License Amendment Request, Revision 1 to the Technical Specifications Conversion to NUREG-1432, Revision 5

References:

1. Florida Power & Light Company (FPL) Letter L-2021-159 dated September 15, 2021, License Amendment Request for the Technical Specifications Conversion to NUREG-1432 Revision 5 (ADAMS Accession No. ML21265A285)
2. St. Lucie Plant, Units 1 and 2 License Amendment Request for the Technical Specifications to NUREG-1432 Revision 5 (NRC ADAMS Package Accession No. ML21265A284)
3. NRC Letter dated November 5, 2021, Notice of Forthcoming Meeting with NextEra/Florida Power and Light (FPL) Regarding the Conversion of Turkey Point Nuclear Generating Station, Units 3 and 4, and St. Lucie Plant, Units 1 and 2 to Improved Technical Specifications (ADAMS Accession No. ML21312A082)
4. NRC Letter dated November 17, 2021, Public Meeting, Turkey Point and St. Lucie Improved Technical Specifications Conversions, Issues Identified to Date – Draft (ADAMS Accession No. ML21320A261)
5. NRC Letter dated December 21, 2021, St. Lucie Plant, Unit Nos. 1 And 2 – Supplemental Information Needed for Acceptance of Licensing Amendment Request Concerning Technical Specification Conversion to NUREG-1432, Revision 5 (EPID L-2021-LLI-0000) (ADAMS Accession Nos. ML21350A426)
6. FPL Letter L-2022-009 dated January 19, 2022, License Amendment Request for the Technical Specifications Conversion to NUREG-1432 Revision 5 – Request for Supplemental Information (RSI) Response (ADAMS Accession No. ML22019A069)
7. NRC Letter dated February 23, 2022, St. Lucie Nuclear Plant, Unit Nos. 1 And 2 – Acceptance of Licensing Amendment Request for The Technical Specification Conversion to NUREG-1432, Revision 5 (EPID L-2021-LLI-0000) (ADAMS Accession Nos. ML22049A553)

FPL, licensee for St. Lucie Plant (PSL), Units 1 and 2, has submitted an Improved Technical Specifications (ITS) conversion License Amendment Request (LAR) for the Facility Operating Licenses for PSL Units 1 and 2 (References 1 and 2). On November 17, 2021, the NRC held a public meeting (References 3) with FPL to discuss review topics of interest (Reference 4) for the ITS conversion LAR submittals. Based on this meeting, the NRC subsequently issued to FPL its request for supplemental information (RSI) for the PSL ITS conversion LAR (Reference 5). FPL responded to the RSI on January 19, 2022 (Reference 6). On February 23, 2022, the NRC issued to FPL the St. Lucie Nuclear Plant, Units 1 and 2 - Acceptance of License Amendment Request for the Technical Specifications Conversion to NUREG-1432, Revision 5 (EPID L-2021-LLI-0000).

This submittal incorporates the changes as identified in Reference 6 and other enhancements. Enhancements include self-identified editorial and typographical corrections and as identified by the

NRC in Enclosure 2 of Reference 5. These enhancements support the goal of conforming, to the extent practical, to Revision 5 of NUREG-1432.

There are six enclosures to this letter. Enclosures 2 and 4 contain content changes. There are no changes to Enclosures 1, 3, 5, and 6. Revisions in Enclosures 2 and 4 revisions are denoted by revision bars. Enclosures 1, 3, 5, and 6 are included in Revision 1 for completeness.

In addition to the enclosures, the letter contains one attachment. Attachment 1 provides a discussion of the LAR Revision 1 content and includes a description of the additional pages added to the ITS Conversion LAR and FPL responses to NRC recommended editorial changes not incorporated.

For ease of reference, the index of enclosed Revision 1 information is provided on page 3 of this letter.

Based on the information provided in Enclosure 2, FPL determined that the proposed license amendment does not involve a significant hazards consideration pursuant to 10 CFR 50.92(c) and that the changes are exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

Should you have any questions regarding this submittal, please contact me at (561) 304-6256 or William.Maher@fpl.com.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 3rd day of June 2022.

Sincerely,

William Maher

Digitally signed by William Maher
DN: cn=William Maher, o=Nuclear, ou=Nuclear Licensing Projects, email=william.maher@fpl.com, c=US
Date: 2022.06.03 10:45:48 -04'00'

William D. Maher
Licensing Director - Nuclear Licensing Projects

cc: Regional Administrator, USNRC, Region II
Project Manager, USNRC, St. Lucie Plant
Senior Resident Inspector, USNRC, St. Lucie Plant
Chief, Bureau of Radiation Control, Florida Department of Health

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ATTACHMENT 1

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CONTENTS OF THE ST. LUCIE PLANT, UNITS 1 AND 2, IMPROVED TECHNICAL SPECIFICATIONS (ITS) REVISION 1 SUBMITTAL

Summary of LAR Changes

By letter dated September 15, 2021 (Reference 1), Florida Power & Light Company (FPL) requested a license amendment to convert St. Lucie Plant Units 1 and 2 (PSL) current Technical Specifications (CTS) to a version based on U.S. NRC NUREG 1432, "Standard Technical Specifications – Combustion Engineering Plants," Revision 5.0 (ISTS) designated as PSL's improved Technical Specifications (ITS). On December 21, 2021 (Reference 2), the NRC provided a request for supplemental information (RSI) to support the NRC staff's acceptance review. The supplemental information requested by the NRC was provided by FPL by letter dated January 19, 2022 (Reference 3). To support the NRC with the formal review of the PSL ITS Conversion license amendment request (LAR), the PSL ITS LAR has been revised to incorporate changes as identified in the FPL response to the NRC RSI during LAR acceptance review.

Revision 1 to the PSL ITS LAR includes:

1. Editorial and typographical corrections self-identified post ITS LAR submittal and as identified by the NRC staff during acceptance review of the ITS application (Reference 3);
2. Incorporation of changes, as applicable, identified in the responses to the NRC RSI (Reference 3);
3. Incorporation of recently revised Unit 1 CTS 3.8.2.1 and Unit 2 CTS 3.8.3.1 pages to reflect PSL Amendments 252 (Unit 1) and 207 (Unit 2) associated with the addition of Risk Informed Completion Times for the Inverters and 120 V AC instrument buses (Reference 4);
4. Modification of noted allowance in ITS 3.4.8, “RCS Loops - MODE 5, Loops Not Filled,” to restore PSL current licensing basis of 1 hour to remove shutdown cooling (SDC) pumps from operation instead of the more restrictive ITS requirement of 15 minutes to remove SDC pumps from operation when switching from one loop to another; and
5. Added Discussion of Change (DOC) L07 to ITS 3.3.1 to address Unit 2 CTS Table 4.3-1, Functional Unit 15 Channel Functional Test Frequency change from “in accordance with the Surveillance Frequency Control Program” to “Once within 7 days prior to each reactor startup” to be consistent with Unit 1 CTS and the ISTS.

Revision to Enclosures 2 and 4 of the original ITS submittal is provided with revision bars annotating the changes. Enclosures 1, 3, 5, and 6 of the original ITS submittal did not require revision but are provided in LAR Revision 1 for completeness.

NRC Recommended Editorials Not Incorporated

In the NRC to FPL letter dated December 21, 2021 (Reference 2), Enclosure 2 provides a list of editorial issues identified by the NRC staff. As stated in the letter, the items in Enclosure 2 were included for FPL’s information only. These editorial issues have been incorporated into Revision 1 of the ITS Conversion LAR except for the following:

Volume 8 – Section 3.3 (ADAMS Accession No. ML21265A293)

1. On pages 68, 69, 80, and 81/450, Both Units – The ITS say to insert figure 2.2-1, but there does not appear to be adequate space on the page. If another page is added the page

numbering will be incorrect. Also figure 3.3.1-3 on ITS page 3.3.1-12 (taken from CTS figure 2.2-2) is labeled page 1 of 1, but it appears to be page 2 of 2. Figure 2.2-1 appears to be page 1 of 2.

FPL Response

No change is made to the submittal. The CTS Figure inserts listed herein correlate the CTS Figures to the ITS Figure 3.3.1-1, ITS Figure 3.3.1-2, and ITS Figure 3.3.1-3, as applicable. ITS Figure 3.3.1-3, as indicated, will contain two pages.

- CTS Figure 2.2-3 will be ITS Figure 3.3.1-1 (Page 1 of 1),
- CTS Figure 2.2-4 will be ITS Figure 3.3.1-2 (Page 1 of 1),
- CTS Figure 2.2-1 will be ITS Figure 3.3.1-3 (Page 1 of 2), and
- CTS Figure 2.2-2 will be ITS Figure 3.3.1-3 (Page 2 of 2).

Additionally, the ITS page numbering is updated in the clean typed ITS that will be provided prior to issuance of the license amendment and based on the changes identified in the ISTS markups. Repagination is not shown in any ISTS markup.

2. On pages 88 and 128/450 and others, Affects Both Units – The bases refer to Variable Power Level – High Trip as the VPHT. Also, the term is written as Variable Power Level Trip – High. The TS Table 3.3.1-1 does not include the term “Trip.” The bases should be consistent with the TS and the acronym should be reflective of the name of the trip or be deleted. This issue shows up multiple times in the bases.

FPL Response

No change is made to the VHPT acronym as it relates to the Reactor Protection System Instrument Function 1, Variable Power Level – High. The VHPT acronym is consistent with PSL terminology for this Function as shown in the Updated Final Safety Analysis Report (UFSAR).

Additional LAR Content

Revision 1 of the ITS LAR contains a total of 14 additional pages not included in Revision 0. The following provides a list of the additional LAR pages and a discussion of the contents of the additional pages:

Volume 6, ITS Section 3.1 (NRC ADAMS Accession No. ML21265A291)

- Added 1 page - Added cover page for Attachment 9, Relocated/Deleted Current Technical Specifications.”

Volume 10, ITS Section 3.5 (NRC ADAMS Accession No. ML21265A295)

- Added 12 pages - Added Attachment 6, ISTS Not Adopted, which indicates ISTS 3.5.5, “Trisodium Phosphate (TSP),” is not adopted for the PSL Unit 1 ITS. TSP is not included in the PSL Unit 1 design and licensing basis for removal of iodine fission product inventory and control of pH in the containment sump during accident conditions. PSL Unit 1 utilizes the Spray Additive System to remove iodine fission product inventory and control pH in the containment sump during accident conditions.

Volume 12, ITS Section 3.7 (NRC ADAMS Accession No. ML21265A297) - Attachment 18, Relocated/Deleted Current Technical Specifications

- Deleted 26 pages – Removed pages that were partial duplicate of ITS 3.7.11.

- Added 15 pages - 8 pages added for relocation of Unit 1 and Unit 2 CTS 3/4.7.2, Steam Generator Pressure/Temperature and 7 pages added for relocation of Unit 2 CTS 3/4.7.6, Flood Protection.
- Total reduction of 11 pages in Section 3.7 from Volume 12 of Enclosure 2 of the ITS LAR Rev. 0.

Volume 13, ITS Section 3.8 (NRC ADAMS Accession No. ML21265A298)

- ITS 3.8.5 - Discussion of Changes – Added 1 page. Added DOC L05 to ITS 3.8.5, DC Sources, Discussion of Changes which changes the number of pages from 4 pages to 5 pages.

Volume 14, ITS Section 3.9 (NRC ADAMS Accession No. ML21265A299)

- Added Attachment 7, Relocated/Deleted Current Technical Specifications which added 9 pages for relocation of Unit 1 and Unit 2 CTS 3/4.9.3, Decay Time.

Volume 16, ITS Chapter 5.0 (NRC ADAMS Accession No. ML21265A301)

- ITS Section 5.2 - Discussion of Changes - Added 1 page. Added DOC A06 to ITS 5.2, Organization, which changes the number of pages from 4 pages to 5 pages.

- ITS Section 5.5 - Discussion of Changes - Added 1 page. Added DOC M04 and DOC LA08 to ITS 5.5, Programs and Manuals, Discussion of Changes which changes the number of pages from 11 pages to 12 pages.

References

1. Florida Power & Light Company (FPL) Letter L-2021-159 dated September 15, 2021, “License Amendment Request for the Technical Specifications Conversion to NUREG-1432 Revision 5.” (ADAMS Accession No. ML21265A285)
2. NRC letter dated December 21, 2021, “St. Lucie Plant, Unit Nos. 1 and 2 – Supplemental Information Needed for Acceptance of Licensing Amendment Request Concerning Technical Specification Conversion to NUREG-1432, Revision 5 (EPID L-2021-LLI-0000).” (ADAMS Accession No. ML21350A426)
3. FPL Letter L-2022-009 dated January 19, 2022, “License Amendment Request for the Technical Specifications Conversion To NUREG-1432 Revision 5 – Request For Supplemental Information (RSI) Response.” (ADAMS Accession No. ML22019A069)
4. NRC letter dated January 14, 2022, “St. Lucie Nuclear Plant, Unit Nos. 1 and 2 – Issuance of Amendment Nos. 252 and 207 to Allow Risk-Informed Completion Times (RICT) for the 120-Volt Alternating Current (AC) Instrument Bus Requirements (EPID L-2020-LLA-0283).” (ADAMS Accession No. ML21342A209)

ENCLOSURE 1

CONTENTS OF THE ST. LUCIE PLANT, UNITS 1 AND 2 IMPROVED TECHNICAL SPECIFICATIONS (ITS) SUBMITTAL

(9 TOTAL PAGES, INCLUDING COVER SHEETS)

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CONTENTS OF THE ST. LUCIE PLANT IMPROVED TECHNICAL SPECIFICATIONS (ITS) SUBMITTAL

Executive Summary

The proposed changes from the St. Lucie Plant (PSL) Unit 1 and Unit 2 current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) are based on Revision 5.0 of NUREG 1432, “Standard Technical Specifications – Combustion Engineering Plants.” There are no additional NRC approved generic changes (Technical Specification Task Force (TSTF) change travelers) since approval of NUREG-1432, Revision 5, to address in this license amendment request as of September 15, 2021. Additionally, there are no linked PSL submittals associated with this ITS license amendment request. Technical Specification Task Force (TSTF)-GG-13-01, “Improved Technical Specifications Conversion Guidance,” Revision 0, was used to prepare the content of the license amendment request.

- The Battery Monitoring and Maintenance Program is included to provide for battery restoration and maintenance per the guidance of IEEE 450-2010, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications.” The Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (76FR54510) references the model application and safety evaluation for plant-specific adoption of TSTF–500, Revision 2 (NRC ADAMS Accession No. ML111751792). PSL has verified the applicable information specified in Section 2.2 of the TSTF-500 model application, including applicable UFSAR information. FPL has verified the following information in support of PSL adoption of TSTF-500, Revision 2:

- A letter has been obtained from the manufacturer of the batteries used at PSL Units 1 and 2 verifying the acceptability of using float current monitoring instead of specific gravity monitoring as a reliable and accurate indication of the state-of-charge of the battery and that this will hold true over the life of the battery.

- Measurement equipment used to monitor float current has the necessary accuracy and capability to measure electrical currents in the expected range.

- Spare battery chargers are available and appropriately sized to support each battery. Each 125 VDC battery charger is a full capacity charger. Typically, two battery chargers are operated in parallel to supply the battery and carry the DC loads. However, either connected battery charger can meet full DC bus load demand and recharge its associated battery. In addition, an additional full capacity swing battery charger can be connected to either DC bus.

- Maintaining battery connection resistances below the battery manufacturer specifications keeps the battery bounded by the battery performance data that was used to qualify the cells. The connection resistance readings are maintained within

20% of the baseline resistance values determined during installation for optimum performance of the battery system. IEEE 450-2010 supports this methodology.

Commitments associated with TSTF-500 are provided in Enclosure 5. A description of the UFSAR changes associated with TSTF-500 are provided in Enclosure 6. Commitments and UFSAR revision packages associated with TSTF-500 will be completed prior to implementation of the ITS Amendment.

Several risk informed initiatives incorporated into the Improved Standard Technical Specifications (ISTS) and associated Bases specified in NUREG-1432, Rev. 5.0, have been previously approved and incorporated into the PSL Unit 1 and Unit 2 Technical Specifications. Therefore, information related to the NRC reviewer notes provided in the ISTS and ISTS Bases associated with these risk informed initiatives are provided in the LAR submittals, FPL responses to NRC staff requests for additional information, and accompanying safety evaluations associated with these risk informed license amendments.

PSL adopted a Surveillance Frequency Control Program (SFCP) on June 22, 2015, in License Amendments 223 and 173, for Unit 1 and Unit 2 respectively (NRC ADAMS Accession No. ML15127A066). In the NRC safety evaluation accompanying the SFCP amendments, the NRC concluded that the adoption of TSTF-425, Revision 3, and risk-informed methodology of NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1, as referenced in the Administrative Controls section of the Technical Specifications, satisfies the key principles of risk-informed decisionmaking applied to the relocated Surveillance frequencies as delineated in Regulatory Guide 1.177, Revision 1, and Regulatory Guide 1.174, Revision 2. Changes to Surveillance frequencies listed in the SFCP are made in accordance with NEI 04-10, Revision 1, as specified in CTS Chapter 6 (ITS Chapter 5).

PSL adopted TSTF-422, "Change in Technical Specifications End States (CE NPSD-1186)," Revision 2, with application of site-specific variations and deviations from TSTF-422, on August 30, 2016, in License Amendments 234 and 184, for Unit 1 and Unit 2 respectively (NRC ADAMS Accession No. ML16210A374). As stated in several NRC reviewer notes in the ISTS Bases, the adoption of a MODE 4 end state requires the licensee to make commitments to follow the guidance established in Section 11 of NUMARC 93-01, “Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Nuclear Management and Resource Council, Revision [4F] and in Revision 2 of WCAP-16364-NP, “Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422).” FPL conducts risk assessments using the procedures and guidance endorsed by Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Regulatory Guide 1.160 endorses the guidance in Section 11 of NUMARC 93-01, Revision 4A. As stated in the safety evaluation accompanying the TSTF-422 End-State amendments, FPL also reviewed the supporting Topical Report WCAP-16364-NP, Revision 2. The NRC staff concluded that PSL's commitment to Regulatory Guide 1.160 guidance is acceptable for application of TSTF-422.

PSL adopted TSTF -505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b" to include Risk Informed Completion Times (RICTs) for

selected Technical Specification systems. License Amendments 247 and 199, were issued on July 2, 2019, for Unit 1 and Unit 2 respectively, to incorporate the RICTs and an associated administrative program (NRC ADAMS Accession No. ML19113A099). In the NRC safety evaluation accompanying the RICT amendment, the NRC found that the PSL PRA maintenance and change process ensures that the configuration risk management program models used in the RICT calculations will continue to use PRA methods acceptable to the NRC and that the PRA model will be updated as necessary to reflect the as-built and as-operated plant. The NRC also found that appropriate programmatic and procedural controls for the RICT Program are consistent with the guidance of NEI 06-09, Revision 0-A. Controls to calculate Risk Informed Completion Times (RICTs) are provided in the CTS Chapter 6 (ITS Chapter 5) and implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The LAR submittal, FPL responses to NRC staff requests for additional information, and safety evaluation accompanying the RICT amendment address the implementation of the RICT program to perform a contemporaneous assessment of the overall impact on safety of proposed plant configurations prior to performing and during performance of maintenance activities that remove equipment from service as stated in several NRC reviewer notes in the ISTS Bases.

Enclosure 2 of the submittal for the conversion of the PSL Unit 1 and Unit 2 CTS to the ITS provides the proposed changes to the CTS, discussion of changes for the proposed CTS changes, the PSL Unit 1 and Unit 2 ITS and ITS Bases, as marked up utilizing NUREG-1432, and justification for deviations from the NUREG. Enclosure 2 consists of the following sixteen volumes:

Volume Titles

1. Application of Selection Criteria to the St. Lucie Plant Technical Specifications
2. Generic Determination of No Significant Hazards Considerations and Environmental Assessment
3. ITS Chapter 1.0, Use and Application
4. ITS Chapter 2.0, Safety Limits
5. ITS Section 3.0, Limiting Condition for Operation (LCO) Applicability and Surveillance Requirement (SR) Applicability
6. ITS Section 3.1, Reactivity Control Systems
7. ITS Section 3.2, Power Distribution Limits
8. ITS Section 3.3, Instrumentation
9. ITS Section 3.4, Reactor Coolant System
10. ITS Section 3.5, Emergency Core Cooling Systems (ECCS)
11. ITS Section 3.6, Containment Systems
12. ITS Section 3.7, Plant Systems
13. ITS Section 3.8, Electrical Power Systems
14. ITS Section 3.9, Refueling Operations
15. ITS Chapter 4.0, Design Features
16. ITS Chapter 5.0, Administrative Controls

Volume 1 is provided to assist the Nuclear Regulatory Commission (NRC) in the review and approval of Volumes 2 through 16. Below is a brief description of the content of each of the volumes in this submittal.

Volume 1

Volume 1 provides details concerning the application of the selection criteria to the individual PSL Unit 1 and Unit 2 CTS. Each CTS Specification is evaluated, and a determination is made as to whether the CTS Specification meets the criteria in 10 CFR 50.36(c)(2)(ii) for retention in the proposed ITS.

Volume 2

Volume 2 contains generic evaluations required by 10 CFR 50.91(a), which support a finding of No Significant Hazards Consideration (NSHC). Based on the inherent similarities in the NSHC evaluations, generic evaluations for a finding of NSHC have been provided for the following categories of CTS changes:

- Administrative Changes
- More Restrictive Changes
- Relocated Specifications
- Removed Detail Changes
- Less Restrictive Changes - Category 1 - Relaxation of LCO Requirements
- Less Restrictive Changes - Category 2 - Relaxation of Applicability
- Less Restrictive Changes - Category 3 - Relaxation of Completion Time
- Less Restrictive Changes - Category 4 - Relaxation of Required Action
- Less Restrictive Changes - Category 5 - Deletion of Surveillance Requirement
- Less Restrictive Changes - Category 6 - Relaxation of Surveillance Requirement Acceptance Criteria
- Less Restrictive Changes - Category 7 - Relaxation of Surveillance Frequency
- Less Restrictive Changes - Category 8 - Deletion of Surveillance Requirement Shutdown Performance Requirements
- Less Restrictive Changes - Category 9 - Deletion of Reporting Requirements

For less restrictive changes not covered by a generic Less Restrictive Change category, specific NSHC evaluations have been provided in the applicable Chapter, Section, or Specification in Volumes 3 through 16. Additionally, specific NSHC evaluations have been provided for changes that have been identified beyond the CTS and the ISTS consistent with the guidance of TSTF-GG-13-01.

In addition, Volume 2 contains an evaluation of environmental consideration in accordance with 10 CFR 51.21. It has been determined that the proposed license amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(b), and no environmental impact statement or environmental assessment need be prepared in connection with the proposed license amendment.

Volumes 3 through 16

Volumes 3 through 16 provide the details and justification to support the proposed changes. Each volume corresponds to a Chapter, Section, or Specification of NUREG-1432. Volumes 3, 4, and 15 are arranged by ITS Chapter. Volumes 5 and 16 are arranged by ITS Section. Volumes 6 through 14 are arranged by ITS Specification. Each volume contains the required information to review the conversion to ITS, and include the following:

- Individual Chapter, Section, or Specification, as applicable, in ITS order.
- Relocated/Deleted CTS Specifications (if applicable); and
- ISTS Specifications not adopted in the PSL Unit 1 and Unit 2 ITS (if applicable).

The information for each Chapter, Section, and Specification, as applicable, is organized as follows:

CTS Markup and Discussion of Changes (DOCs)

This section contains a markup of the CTS pages, either for CTS pages associated with an Individual ITS Specification or for Relocated/Deleted CTS Specifications, and the DOCs from the CTS. Other CTS license amendment requests under NRC review, as described in Enclosure 4 of this submittal, are not linked to this license amendment request and have not been incorporated in the proposed change.

The CTS markup pages for each ITS Specification are normally in numerical order. However, more than one CTS Specification is sometimes used in the generation of an ITS Specification. In this case, the CTS pages that are the major contributor to the ITS Specification are shown first, followed by the remaining associated CTS pages in numerical order.

The left-hand margin of the CTS markup pages includes a cross-reference to the equivalent ITS requirement. The upper right-hand corner of the CTS markup pages is annotated with the ITS Specification number to which it applies. Items on the CTS markup pages that are addressed in other proposed ITS Chapters, Sections, or Specifications are annotated with the appropriate reference.

The CTS markup pages are annotated with an alphanumeric designator to identify the differences between the CTS and the proposed ITS. The designator corresponds to a DOC, which provides the description and justification of the change to the CTS. The DOCs are located directly following the associated CTS markup for each Chapter, Section, or Specification, as applicable.

Each proposed change to the CTS is designated as one of the following DOC categories:

Designator	**Category**
A	ADMINISTRATIVE CHANGES – Changes to the CTS that do not result in new requirements or change operational restrictions or flexibility. These changes are supported in aggregate by a single generic NSHC.
M	MORE RESTRICTIVE CHANGES - Changes to the CTS that result in added restrictions or reduced flexibility. These changes are supported in aggregate by a single generic NSHC.
R	RELOCATED SPECIFICATIONS - Changes to the CTS that relocate specifications that do not meet the selection criteria of

10 CFR 50.36(c)(2)(ii). These changes are supported in aggregate by a single generic NSHC.

LA REMOVED DETAIL CHANGES - Changes to the CTS that eliminate detail and relocate the detail to a licensee controlled document. Typically, this involves details of system design and function, or procedural detail on methods of conducting a Surveillance Requirement. These changes are supported in aggregate by a single generic NSHC. In addition, the generic type of removed detail change is identified in italics at the beginning of the DOC.

L LESS RESTRICTIVE CHANGES - Changes to the CTS that result in reduced restrictions or added flexibility. These changes are supported either in aggregate by a generic NSHC that addresses a particular category of less restrictive change, or by a specific NSHC if the change is not covered by one of the generic categories of less restrictive changes. If the less restrictive change is covered by a generic NSHC, the category of the change is identified in italics at the beginning of the DOC.

The DOCs are numbered sequentially within each letter designator for each ITS Chapter, Section, or Specification. Where a proposed change is applicable to only Unit 1 or Unit 2, it is identified as such at the beginning of the DOC in bold text.

The CTS Bases pages are replaced in their entirety by the proposed PSL Unit 1 and Unit 2 ITS Bases. Therefore, the CTS Bases markup pages are not provided in the ITS submittal.

ISTS Markup and Justification for Deviations (JFDs)

This section contains a markup of the NUREG-1432, Volume 1, ISTS pages, either for ISTS pages associated with an Individual ITS Specification or ISTS Specifications not adopted in the PSL ITS, and JFDs from the ISTS. The ISTS pages are annotated with a numeric designator to identify the differences between the ISTS and the proposed ITS. The designator corresponds to a JFD, which provides the justification for the difference. Where a JFD is applicable to only Unit 1 or Unit 2, it is identified as such at the beginning of the JFD in bold text. The JFDs are located directly following the associated ISTS markup for each Chapter, Section, or Specification, as applicable.

The left-hand margin of the ISTS markup pages includes a cross-reference to the equivalent CTS requirement.

ISTS Bases Markup and JFDs

This section contains a markup of the NUREG-1432, Volume 2, ISTS Bases pages, either for ISTS Bases pages associated with an Individual ITS Specification or ISTS Specifications not adopted in the PSL Unit 1 and 2 ITS, and JFDs from the ISTS Bases. The ISTS Bases pages are annotated with a numeric designator to identify the differences between the ISTS Bases and the proposed ITS Bases. The designator corresponds to a JFD, which provides the justification for the difference. Where a JFD is applicable to only Unit 1 or Unit 2, it is identified as such at the beginning of the JFD

in bold text. The Bases JFDs are located directly following the associated ISTS Bases markup for each Chapter, Section, or Specification, as applicable. The volumes for ITS Chapters 1.0, 4.0, and 5.0 do not include this section, because NUREG-1432 does not include any Bases for these Chapters.

Determination of NSHC

This section contains the determination in accordance with 10 CFR 50.91(a)(1) using the criteria of 10 CFR 50.92(c) to support a finding of NSHC. For changes covered by a generic NSHC, the generic NSHCs are in Volume 2. For less restrictive changes not covered by a generic less restrictive category or have been determined to be a change beyond the CTS and the ITS, a specific NSHC evaluation has been performed. Each evaluation is annotated to correspond to the DOC discussed in the specific NSHC evaluation. For ITS Chapters, Sections, or Specifications for which the changes are covered by a generic NSHC evaluation, a statement that there are no specific NSHCs is provided.

ENCLOSURE 2

ST. LUCIE PLANT, UNITS 1 AND 2
IMPROVED TECHNICAL SPECIFICATIONS (ITS) REVISION 1 SUBMITTAL
VOLUMES 1 THROUGH 16

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VOLUME 1

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

APPLICATION OF SELECTION CRITERIA TO THE ST. LUCIE PLANT UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS

Revision 1

APPLICATION OF SELECTION CRITERIA TO THE ST LUCIE PLANT UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS

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APPLICATION OF SELECTION CRITERIA
TO THE ST LUCIE PLANT UNIT 1 AND UNIT 2
TECHNICAL SPECIFICATIONS

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APPENDIX

A. JUSTIFICATION FOR SPECIFICATION RELOCATION

APPLICATION OF SELECTION CRITERIA
TO THE ST LUCIE PLANT UNIT 1 AND UNIT 2
TECHNICAL SPECIFICATIONS

1. INTRODUCTION

The purpose of this document is to confirm the results of the Combustion Engineering (C-E) Owners Group application of the Technical Specification selection criteria on a plant specific basis for the St. Lucie Nuclear Plant (PSL) Unit 1 and Unit 2. Florida Power & Light Company (hereinafter FPL) has reviewed the application and confirmed the applicability of the selection criteria to each of the Technical Specifications utilized in report CEN-355, "C-E Owners Group Restructured Standard Technical Specifications Volume 1 Criteria Application," (Reference 1), NRC Staff Review of nuclear steam supply system (NSSS) Vendor Owners Groups Application of The Commission's Interim Policy Statement Criteria To Standard Technical Specifications, Wilgus/Murley letter dated May 9, 1988 (Reference 2), and as revised in NUREG-1432, Revision 5.0 "Standard Technical Specifications, Combustion Engineering Plants" (Reference 3) and applied the criteria to each of the current PSL Technical Specifications. Additionally, in accordance with the NRC Final Policy Statement (Reference 4), this confirmation of the application of selection criteria includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in Reference 1, as applicable to the PSL.

2. SELECTION CRITERIA

FPL has utilized the selection criteria provided in Section 36 of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50.36), dated July 19, 1995 (Reference 5), which were codified following issuance of the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (Reference 4) to develop the results contained in the attached matrix. PRA insights as used in the C-E Owners Group submittal were utilized, confirmed by FPL, and are discussed in the next section of this report. The selection criteria and discussion provided in Reference 4 are as follows:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary:

Discussion of Criterion 1: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident.

This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion should not, however, be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 2: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing DBA and transient analyses and that the plant will be operated to preclude unanalyzed transients and accidents. These analyses consist of postulated events, analyzed in

the Final Safety Analysis Report (FSAR), for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the FSAR (or equivalent chapters) and are identified as Condition II, III, or IV events (ANSI N18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

As used in Criterion 2, process variables are only those parameters for which specific values or ranges of values have been chosen as reference bounds in the DBA or transient analyses and which are monitored and controlled during power operation such that process values remain within the analysis bounds. Process variables captured by Criterion 2 are not, however, limited to only those directly monitored and controlled from the control room.

These could also include other features or characteristics that are specifically assumed in DBA and transient analyses even if they cannot be directly observed in the control room (e.g, moderator temperature coefficient and hot channel factors).

The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. If these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low. This criterion also includes active design features (e.g., high pressure/low pressure system valves and interlocks) and operating restrictions (pressure/temperature limits) needed to preclude unanalyzed accidents and transients.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated DBA or transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequences of the DBA or transient. Safety sequence analyses or their equivalent have been performed in

recent years and provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's DBA and transient analyses, as presented in Chapters 6 and 15 of the plant's Final Safety Analysis Report (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria.

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path for a particular mode of operation does not include backup and diverse equipment (e.g., rod withdrawal block which is a backup to the average power range monitor high flux trip in the startup mode, safety valves which are backup to low temperature overpressure relief valves during cold shutdown).

Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety:

Discussion of Criterion 4: It is the Commission policy that pressurized water reactor (PWR) licensees retain in their Technical Specifications LCOs, action statements and Surveillance Requirements for the Residual Heat Removal System, which operating experience and PRA have generally shown to be significant to public health and safety, and any other structures, systems, or components that meet this criterion.

The Commission recognizes that other structures, systems, or components may meet this criterion. Plant and design-specific PRA's have yielded valuable insight to unique plant vulnerabilities not fully recognized in the safety analysis report DBA or transient analyses. It is

the intent of this criterion that those requirements that PRA or operating experience exposes as significant to public health and safety, consistent with the Commission's Safety Goal and Severe Accident Policies, be retained or included in Technical Specifications.

The Commission expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant specific PRA or risk survey and any available literature on risk insights and PRAs. This material should be employed to strengthen the technical bases for those requirements that remain in Technical Specifications, when applicable, and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Similarly, the NRC staff will also employ risk insights and PRAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

3. PRA INSIGHTS

Introduction and Objectives

Reference 4 includes a statement that NRC expects licensees to utilize any plant specific PRA or risk survey and any available literature on risk insights and PRAs to strengthen the technical bases for these requirements that remain in Technical Specifications and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Those Technical Specifications proposed as being relocated to other plant controlled documents will be maintained under programs subject to the 10 CFR 50.59 review process. These Relocated Specifications have been compared to a variety of PRA material with two purposes: 1) to identify if a Specification component or topic is addressed by PRA; and 2) if addressed, to judge if the Relocated Specification component or topic is risk-important. The intent of the PRA review was to provide an additional screen to the deterministic criteria. This review was accomplished in the generic C-E Owners Group submittal CEN-355 (Reference 1). The results

of this generic review have been confirmed by FPL for the applicable PSL Specifications to be relocated.

Assumptions and Approach

The results of the generic criteria application documented in CEN-355 were reviewed to identify systems and components to be relocated from the Technical Specifications to other means of control. This list was compared to the findings from several PRAs to determine if any of the items proposed to be removed from the Technical Specifications were risk significant.

At the time CEN-355 was issued, the only published PRA for a C-E designed NSSS was the Calvert Cliffs Unit 1 Interim Reliability Evaluation Program (IREP) (NUREG/CR-3511, “Interim Reliability Evaluation Program: Analysis of the Calvert Cliffs Unit 1 Nuclear Power Plant," dated August 1984). This IREP was examined to identify the systems and components in the risk dominant event sequences. Event sequences with lower frequency of occurrence than the cutoff of 1.0 E-6 per year were also reviewed to ensure that systems of prime importance were not overlooked. Tables 3-1 and 3-2 of CEN-355 summarized the results of the examination of the risk significant event sequences. As noted in CEN-355, Calvert Cliffs Unit 1 is of the same design as PSL Units 1 and 2.

Table 3-1 represented those systems and components found to be risk significant; i.e., systems and components required to terminate or mitigate the event sequences leading to core melt having a frequency of occurrence of at least 1.0 E-6 per year.

Table 3-2 identified the dominant accident sequences identified as leading to core melt which have a frequency greater than E-6 per year.

To ensure an adequate representation of plants with a C-E designed NSSS, several partial PRAs for plants with a C-E NSSS were examined. These included:

- CEN-239, “Probabilistic Risk Assessment of the Effect of PORVs on Depressurization and Decay Heat Removal,” Supplements 1, 2, and 3, June 1983;
- PRA evaluations of Palisades main steam line break events; and

- PSL Shutdown Decay Heat Removal Analysis NUREG/CR-4710, “Shutdown Decay Heat Removal Analysis, Combustion Engineering 2-Loop PWR Case Study,” 1987.

Also, full PRAs on PWRs with other NSSS designs were examined. These included:

- Oconee, a two loop Babcock and Wilcox PWR, and
- Seabrook, a four loop Westinghouse PWR.

To ensure that no risk-significant systems were overlooked, a listing of the requirements not meeting criteria for retention as technical specifications was reviewed and documented in Table 3-3 of CEN-355, Appendix B.

Appendix C of CEN-355 includes safety sequence diagrams for identified types of initiating events considered in formal risk assessments. Table C 1 in Appendix C, CEN-355 tabulated the events for which the safety sequence analysis (SSA) were performed. The results of the SSA are summarized in the sequence of events diagrams provided in Table C.1 of Appendix C, CEN-355.

4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 herein were applied to the PSL Unit 1 and Unit 2 Technical Specifications. The following Summary Disposition Matrix for St. Lucie Unit 1 and Unit 2 is a summary of that application indicating which Specifications are being retained or relocated, the criteria for inclusion, if applicable, the NRC results of the criteria application as expressed in the NRC Staff Review of NSSS Vendor Owners Groups Application of The Commission's Interim Policy Statement Criteria To Standard Technical Specifications, Wilgus/Murley letter dated May 9, 1988 (Reference 2), and any necessary explanatory notes. Discussions that document the rationale for the relocation of each Specification which failed to meet the selection criteria are provided in Appendix A, except as noted in the Summary Disposition Matrix.

5. REFERENCES

1. CEN-355, "C-E Owners Group Restructured Standard Technical Specifications Volume 1 Criteria Application," December 1987.

2. Letter from T.E. Murley (NRC) to W.S. Wilgus (B&W Owners Group), "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application Of The Commission's Interim Policy Statement Criteria To Standard Technical Specifications," dated May 9, 1988.

3. NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 5.0, September 2021.

4. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (Federal Register Notice 58 FR 39132).

5. 10 CFR 50.36, "Technical specifications," (c)(2)(ii) selection criteria, July 19, 1995 (Federal Register Notice 60 FR 36953).

ATTACHMENT 1

SUMMARY DISPOSITION MATRIX FOR ST LUCIE UNIT 1 AND UNIT 2

SUMMARY DISPOSITION MATRIX FOR ST. LUCIE PLANT (PSL) UNIT 1 AND UNIT 2

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
1.0		**DEFINITIONS**	1.1	Yes	This section provides definitions for several defined terms used throughout the remainder of Technical Specifications. They are provided to improve the meaning of certain terms. As such, direct application of the Technical Specification selection criteria is not appropriate. However, only those definitions for defined terms that remain because of application of the selection criteria, will remain as definitions in this section of Technical Specifications.
2.1		**SAFETY LIMITS**			
2.1.1		Reactor Core	2.1.1	Yes	Application of Technical Specification selection criteria is not appropriate. However, Safety Limits will be included in Technical Specifications as required by 10 CFR 50.36.
2.1.2		Reactor Coolant System Pressure	2.1.2	Yes	Same as above.

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
2.2		**LIMITING SAFETY SYSTEM SETTINGS**			
2.2.1		Reactor Trip Setpoints except Unit 2 Table 2.2-1, Functional Unit 10	3.3.1	Yes-3	The Reactor Protection System (RPS) limited safety system settings (LSSS) have been included as part of the RPS instrumentation Specification, which has been retained since the Functions either actuate to mitigate consequences of design basis accidents and transients or are retained as directed by the NRC as the Functions are part of the RPS.
NA	2.2.1	Reactor Protective Instrumentation Trip Setpoint Limits Unit 2 Table 2.2-1, Functional Unit 10	Relocated	No	See Appendix A, page 1.
3/4.0		**APPLICABILITY**			
3.0.1		Limiting Condition For Operation	LCO 3.0.1	Yes	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1432, Revision 5.

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3.0.2		Limiting Condition For Operation	LCO 3.0.2	Yes	Same as above.
3.0.3		Limiting Condition For Operation	LCO 3.0.3	Yes	Same as above.
3.0.4		Limiting Condition For Operation	LCO 3.0.4	Yes	Same as above.
3.0.5		Limiting Condition For Operation	LCO 3.0.5	Yes	Same as above.
3.0.6		Limiting Condition For Operation	LCO 3.0.6	Yes	Same as above.
4.0.1		Surveillance Requirements	SR 3.0.1	Yes	Same as above.
4.0.2		Surveillance Requirements	SR 3.0.2	Yes	Same as above.
4.0.3		Surveillance Requirements	SR 3.0.3	Yes	Same as above.
4.0.4		Surveillance Requirements	SR 3.0.4	Yes	Same as above.
4.0.5		Surveillance Requirements	Deleted	No	Deleted. See CTS 4.0.5 technical change discussion in the Discussion of Changes for ITS Section 3.0.
3/4.1		**REACTIVITY CONTROL SYSTEMS**			
3/4.1.1		*BORATION CONTROL*			

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.1.1.1		Shutdown Margin – Tavg > 200°F	3.1.2 3.1.4 3.1.5 3.1.6	Yes-2	
3/4.1.1.2		Shutdown Margin – Tavg ≤ 200°F	3.1.1	Yes-2	
3/4.1.1.3		Boron Dilution	Relocated	No	See Appendix A, page 3.
3/4.1.1.4		Moderator Temperature Coefficient	3.1.3	Yes-2	
3/4.1.1.5		Minimum Temperature for Criticality	3.4.2	Yes-2	
3/4.1.2		*BORATION SYSTEMS*			
3/4.1.2.1		Flow Paths – Shutdown	Relocated	No	See Appendix A, page 4.
3/4.1.2.2		Flow Paths – Operating	Relocated	No	See Appendix A, page 4.
3/4.1.2.3		Charging Pumps – Shutdown	Relocated	No	See Appendix A, page 4.
3/4.1.2.4		Charging Pumps – Operating	Relocated	No	See Appendix A, page 4.
3/4.1.2.5		Boric Acid Pumps – Shutdown	Relocated	No	See Appendix A, page 4.
3/4.1.2.6		Boric Acid Pumps – Operating	Relocated	No	See Appendix A, page 4.

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.1.2.7		Borated Water Sources – Shutdown	Relocated	No	See Appendix A, page 4.
3/4.1.2.8		Borated Water Sources – Operating	Relocated	No	See Appendix A, page 4.
3/4.1.3		*MOVABLE CONTROL ASSEMBLIES*			
3/4.1.3.1		[Full Length - Unit 1] CEA Position	3.1.4	Yes-2, 3	
3/4.1.3.3	3/4.1.3.2	Position Indicator Channels [- Operating - Unit 2]	3.1.7	Yes-2	Based on NUREG-1431 (Westinghouse STS) ISTS 3.1.7
NA	3/4.1.3.3	**Unit 2 only -** Position Indicator Channels - Shutdown	Relocated	No	See Appendix A, page 6
3/4.1.3.4		CEA Drop Time	3.1.4	Yes-2	
3/4.1.3.5		Shutdown CEA Insertion Limit	3.1.5	Yes-2	
3/4.1.3.6		Regulating CEA Insertion Limits	3.1.6	Yes-2	
3/4.2		**POWER DISTRIBUTION LIMITS**			
3/4.2.1		Linear Heat Rate	3.2.1	Yes-2	
3/4.2.3		Total Integrated Radial Peaking Factor – F^T_r	3.2.2	Yes-2	ISTS 3.2.3
3/4.2.4		Azimuth Power Tilt - Tq	3.2.3	Yes-2	ISTS 3.2.4

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.2.5		DNB Parameters	3.2.4 3.4.1	Yes-2	ISTS 3.2.5 - ASI ISTS 3.4.1 - RCS Flow Rate, RCS Cold Leg Temp, Pressurizer Pressure
3/4.3		**INSTRUMENTATION**			
3/4.3.1		Reactor Protective Instrumentation except Unit 2 Table 3.3-1, Functional Unit 10	3.3.1 3.3.2 3.3.11	Yes-3	ISTS 3.3.1 ISTS 3.3.3 ISTS 3.3.13
NA	3/4.3.1	Reactor Protective Instrumentation Unit 2 Table 3.3-1, Functional Unit 10	Relocated	No	See Appendix A, page 1
3/4.3.2		Engineered Safety Feature Actuation System Instrumentation	3.3.3 3.3.4 3.3.5	Yes-3	ISTS 3.3.4 ISTS 3.3.5 ISTS 3.3.6
3/4.3.3		*MONITORING INSTRUMENTATION*			
3/4.3.3.1		Radiation Monitoring	3.3.6 3.3.7 3.3.8 3.4.15	Yes-3	ISTS 3.3.7 ISTS 3.3.8 ISTS 3.3.10 ISTS 3.4.15
3/4.3.3.5		Remote Shutdown [Instrumentation - Unit 1] [System Instrumentation - Unit 2]	3.3.10	Yes-4	ISTS 3.3.12
3/4.3.3.8	3/4.3.3.6	Accident Monitoring Instrumentation (Type A and non-Type A Category 1 Instruments)	3.3.9	Yes-3, 4	ISTS 3.3.11

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.3.3.8	3/4.3.3.6	Accident Monitoring Instrumentation (Non-Type A and not Category 1 Instruments)	3.3.9	No	See Appendix A, page 7
3/4.4		**REACTOR COOLANT SYSTEM**			
3/4.4.1		*REACTOR COOLANT LOOPS AND COOLANT CIRCULATION*			
3/4.4.1.1		Startup and Power Operation	3.4.4	Yes-2, 3	
3/4.4.1.2		Hot Standby	3.4.5	Yes-3	
3/4.4.1.3		Hot Shutdown	3.4.6	Yes-4	
3/4.4.1.4.1		Cold Shutdown (Loops Filled)	3.4.7	Yes-4	
3/4.4.1.4.2		Cold Shutdown (Loops Not Filled)	3.4.8	Yes-4	
3/4.4.3	3/4.4.2.2	[Safety Valves - Operating - Unit 1] [Operating - Unit 2]	3.4.10	Yes-3	
3/4.4.4	3/4.4.3	Pressurizer	3.4.9	Yes-2, 3	
3/4.4.5		Steam Generator (SG) Tube Integrity	3.4.17	Yes-2	ISTS 3.4.18
3/4.4.6		*REACTOR COOLANT SYSTEM LEAKAGE*			

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.4.6.1		Leakage Detection Systems	3.4.15	Yes-1	
3/4.4.6.2		[Reactor Coolant System - Unit 1] [Operational - Unit 2] Leakage	3.4.13 3.4.14	Yes-2	
3/4.4.8		Specific Activity	3.4.16	Yes-2	
3/4.4.9		*PRESSURE/TEMPERATURE LIMITS*			
3/4.4.9.1		Reactor Coolant System	3.4.3	Yes-2	
3/4.4.9.2		[Pressurizer - Unit 1] [Pressurizer Heatup and Cooldown Limits - Unit 2]	Relocated	No	See Appendix A, page 9.
3/4.4.12	3/4.4.4	PORV Block Valves	3.4.11	Yes-4	
3/4.4.13	3/4.4.9.3	[Power Operated Relief Valves - Unit 1] [Overpressure Protection Systems - Unit 2]	3.4.12	Yes-2	
3/4.4.14	NA	**Unit 1 Only** - Reactor Coolant Pump – Starting	3.4.6 3.4.7	Yes-2	
3/4.4.15	3/4.4.10	Reactor Coolant System Vents	Relocated	No	See Appendix A, page 10.
3/4.5		**EMERGENCY CORE COOLING SYSTEMS (ECCS)**			

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.5.1		Safety Injection Tanks	3.5.1	Yes-3	
3/4.5.2		ECCS Subsystems – Operating	3.5.2 3.5.5 (U2)	Yes-3	
3/4.5.3		ECCS Subsystems – Shutdown	3.5.3	Yes-3	
3/4.5.4		Refueling Water Tank	3.5.4	Yes-3	
3/4.6		**CONTAINMENT SYSTEMS**			
3/4.6.1		*CONTAINMENT VESSEL*			
3/4.6.1.1		[Containment Vessel Integrity – Unit 1] [Containment Integrity – Unit 2]	3.6.1	Yes-3	ISTS 3.6.1B
3/4.6.1.2		Containment Leakage	3.6.1	Yes-3	ISTS 3.6.1B Containment leakage is being retained as a Surveillance Requirement (SR 3.6.1.1) in ITS 3.6.1.
3/4.6.1.3		Containment Air Locks	3.6.2	Yes-3	
3/4.6.1.4		Internal Pressure	3.6.4	Yes-2	ISTS 3.6.4B
3/4.6.1.5		Air Temperature	3.6.5	Yes-2	

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.6.1.6		Containment Vessel Structural Integrity	3.6.1	Yes-3	ISTS 3.6.1B Containment vessel structural integrity is being retained as a Surveillance Requirement (SR 3.6.1.1) in ITS 3.6.1.
NA	3/4.6.1.7	**Unit 2 only** – Containment Ventilation System	3.6.3	Yes-3	ISTS 3.6.3 addresses 48 inch purge and exhaust valves.
3/4.6.2		*DEPRESSURIZATION AND COOLING SYSTEMS*			
3/4.6.2.1		Containment Spray and Cooling Systems	3.6.6	Yes-3	ISTS 3.6.6A
3/4.6.2.2	NA	**Unit 1 only -** Spray Additive System	3.6.10	Yes-3	ISTS 3.6.7
3/4.6.3		Containment Isolation Valves	3.6.3	Yes-3	
3/4.6.5		Vacuum Relief Valves	3.6.8	Yes-3	ISTS 3.6.12
3/4.6.6		*SECONDARY CONTAINMENT*			
3/4.6.6.1		Shield Building Ventilation System	3.6.9	Yes-3	For Unit 2: Combines ISTS 3.6.8 and ISTS 3.7.14
3/4.6.6.2		Shield Building Integrity	3.6.7	Yes-3	ISTS 3.6.11

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.6.6.3		Shield Building Structural Integrity	3.6.7	Yes-3	ISTS 3.6.11 Shield building structural integrity is being retained as a Surveillance Requirement; SR 3.6.7.3 in ITS 3.6.7.
3/4.7		**PLANT SYSTEMS**			
3/4.7.1		*TURBINE CYCLE*			
3/4.7.1.1		Safety Valves	3.7.1	Yes-3	
3/4.7.1.2		Auxiliary Feedwater System	3.7.5	Yes-3	
3/4.7.1.3		Condensate Storage Tank	3.7.6	Yes-2, 3	
3/4.7.1.4		Activity	3.7.1.16	Yes-2	ISTS 3.7.19
3/4.7.1.5		Main Steam Line Isolation Valves	3.7.2	Yes-3	
NA	3/4.7.1.6	Main Feedwater Isolation Valves	3.7.3	Yes-3	Adding ISTS 3.7.3, Main Feedwater Isolation Valves (MFIVs), to Unit 1 ITS.
NA	3/4.7.1.7	Atmospheric Dump Valves	3.7.4	Yes-3	Adding ISTS 3.7.4, Atmospheric Dump Valves (ADVs), to Unit 1 ITS.
3/4.7.2.1	3/4.7.2	Steam Generator Pressure/Temperature Limitation	Relocated	No	See Appendix A, page 11.

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.7.3		Component Cooling Water System	3.7.7	Yes-3	
3/4.7.4		Intake Cooling Water System	3.7.8	Yes-3	
3/4.7.5		Ultimate Heat Sink	3.7.9	Yes-3	
NA	3/4.7.6	**Unit 2 only -** Flood Protection	Relocated	No	See Appendix A, page 12. General requirements associated with hazard barriers, including flooding barriers, that affect Technical Specification systems, trains, subsystems, and components are included in ITS LCO 3.0.9, as modified consistent with NUREG-1432, Revision 5.
3/4.7.7		[Control Room Emergency Ventilation System - Unit 1] [Control Room Emergency Air Cleanup System (CREACS) - Unit 2]	3.7.10 3.7.11	Yes-3	ISTS 3.7.11 ISTS 3.7.12
3/4.7.8		ECCS Area Ventilation System	3.7.12	Yes-3	ISTS 3.7.13
3/4.7.9	3/4.7.10	Sealed Source Contamination	Relocated	No	See Appendix A, page 13.

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.7.10	3/4.7.9	Snubbers	Deleted	No	General requirements associated with snubbers that affect Technical Specification systems, trains, subsystems, and components are included in ITS LCO 3.0.8, as modified consistent with NUREG-1432, Revision 5. See technical change discussions related to Unit 1 CTS 3/4.7.10 and Unit 2 CTS 3/4.7.9 in the Discussion of Changes for ITS Sections 3.0 and 3.7.
3/4.8		**ELECTRICAL POWER SYSTEMS**			
3/4.8.1		*A.C. SOURCES*			
3/4.8.1.1		Operating	3.8.1 3.8.3	Yes-3	
3/4.8.1.2		Shutdown	3.8.2 3.8.3	Yes-3	
NA	*3/4.8.2*	*D.C. SOURCES*			
3/4.8.2.3	3/4.8.2.1	[D.C. Distribution – Operating - Unit 1] [Operating - Unit 2]	3.8.4 3.8.6	Yes-3	
3/4.8.2.4	3/4.8.2.2	[D.C. Distribution – Shutdown - Unit 1] [Shutdown - Unit 2]	3.8.5 3.8.6	Yes-3	

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.8.2	*3/4.8.3*	*ONSITE POWER DISTRIBUTION [SYSTEMS - Unit 1]*			
3/4.8.2.1	3/4.8.3.1	[A.C. Distribution – Operating - Unit 1] [Operating - Unit 2]	3.8.7 3.8.9	Yes-3	Adding ISTS 3.8.7, Inverters - Operating, to Unit 1 ITS
3/4.8.2.2	3/4.8.3.2	[A.C. Distribution – Shutdown - Unit 1] [Shutdown - Unit 2]	3.8.8 3.8.10	Yes-3	Adding ISTS 3.8.7, Inverters - Shutdown, to Unit 1 ITS
3/4.8.2.3	3/4.8.3.1	[D.C. Distribution – Operating - Unit 1] [Operating - Unit 2]	3.8.9	Yes-3	
3/4.8.2.4	3/4.8.3.2	[D.C. Distribution – Shutdown - Unit 1] [Shutdown - Unit 2]	3.8.10	Yes-3	
3/4.9		**REFUELING OPERATIONS**			
3/4.9.1		Boron Concentration	3.9.1	Yes-2	
3/4.9.2		Instrumentation	3.9.2	Yes-3	
3/4.9.3		Decay Time	Deleted	No	See CTS 3/4.9.3 technical change discussion in the Discussion of Changes for ITS Section 3.9.
3/4.9.4		Containment Penetrations	3.9.3	Yes-3	

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.9.8		*SHUTDOWN COOLING AND COOLANT CIRCULATION*			
3/4.9.8.1		High Water Level	3.9.4	Yes-2	
3/4.9.8.2		Low Water Level	3.9.5	Yes-2	
3/4.9.9		Containment Isolation System	3.9.3	Yes-3	
3/4.9.10		Water Level - Reactor Vessel	3.9.6	Yes-2	
3/4.9.11		Spent Fuel Storage Pool	3.7.13 3.7.14	Yes-2, 3 Yes-2	ISTS 3.7.16 ISTS 3.7.17
3/4.9.12	NA	**Unit 1 only -** Fuel Pool Ventilation System - Fuel Storage	3.7.17	Yes-3	ISTS 3.7.14
3/4.10		**SPECIAL TEST EXCEPTIONS**			
3/4.10.1		Shutdown Margin	Deleted	No	Compliance with an STE is optional; therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. See CTS 3/4.10.1 technical change discussion in the Discussion of Changes for ITS Section 3.1.

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
3/4.10.2		[Moderator Temperature Coefficient – Unit 2] Group Height, Insertion and Power Distribution Limits	3.1.8	Yes	Compliance with an STE is optional; therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. This Specification has been retained to provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs.
NA	3/4.10.3	**Unit 2 only** – Reactor Coolant Loops	Deleted.	No	Compliance with an STE is optional; therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. See U2 CTS 3/4.10.3 technical change discussion in the Discussion of Changes for ITS Section 3.4.
3/4.10.5	3/4.10.4	Center CEA Misalignment	Deleted	No	Compliance with an STE is optional; therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. See U1 CTS 3/4.10.5 and U2 CTS 3/4.10.4 technical change discussion in the Discussion of Changes for ITS Section 3.1.

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
NA	3/4.10.5	**Unit 2 only** – CEA Insertion During ITC, MTC, and Power Coefficient Measurements	3.1.8	Yes	Not explicitly included in NUREG-1432. ISTS 3.1.8 limits power to 85% RTP. Compliance with an STE is optional; therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. This Specification has been retained to provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs.
3/4.11		**RADIOACTIVE EFFLUENTS**			
3/4.11.2.5		Explosive Gas Mixture	5.5	Yes	Although this Specification does not meet any Technical Specification selection criteria, it has been retained in accordance with the NRC letter from W. T. Russell to the industry ITS Chairpersons, dated October 25, 1993.
3/4.11.2.6		Gas Storage Tanks	5.5	Yes	Although this Specification does not meet any Technical Specification selection criteria, it has been retained in accordance with the NRC letter from W. T. Russell to the industry ITS Chairpersons, dated October 25, 1993.

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

U1 Current TS (CTS) Number	U2 CTS Number (if different)	CTS Specification Title	New TS (ITS) Reference	Retained/ Criterion For Inclusion (1)	Notes
5.0		**DESIGN FEATURES**	3.7.15 4.0	Yes-2 Yes	ISTS 3.7.18 Application of Technical Specification selection criteria is not appropriate. However, specific portions of Design Features will be included in Technical Specifications as required by 10 CFR 50.36.
6.0		**ADMINISTRATIVE CONTROLS**	5.0	Yes	Application of Technical Specification selection criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.

(1) The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

APPENDIX A

JUSTIFICATION FOR SPECIFICATION RELOCATION

APPLICATION OF SELECTION CRITERIA
TO THE ST LUCIE PLANT UNIT 1 AND UNIT 2
TECHNICAL SPECIFICATIONS

Appendix A – Justification for Specification Relocation

Unit 2 CTS Tables 3.3-1 and 4.3-1, Reactor Protective System Instrumentation, Functional Unit 10
Unit 2 CTS Table 2.2-1, Reactor Protective System Instrumentation Trip Setpoint Limits, Functional Unit 10 | R1

DISCUSSION:

The Reactor Protective System (RPS) initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary (RCPB) during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents. The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as Limiting Conditions for Operation on other reactor system parameters and equipment performance. Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions.

The NRC position on application of the screening criteria to RPS instrumentation is documented in a letter dated May 9, 1988 from T.E. Murley (NRC) to the Owners Groups (Reference 2). The Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function specified in CTS Table 2.2-1, 3.3-1 and 4.3-1 is provided as a plant specific reactor trip as an equipment protective feature required as a stipulation by NRC staff in justifying acceptance of the Reactor Coolant Water Component Cooling Water piping design. This reactor trip is not required for reactor protection and is not assumed in any accident or transient analysis. In addition, this automatic reactor trip feature on a loss of component cooling water (CCW) flow to the reactor coolant pumps (RCPs) is not required to protect against violating the core specified acceptable fuel design limits or breaching the RCPB during AOOs and is not considered an LSSS for a variable that has a significant safety function. | R1

COMPARISON TO SELECTION CRITERIA:

1. The Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary.

2. The Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

3. The Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function is not a structure, system, or component that is part of the primary success path or which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The loss of CCW flow to the RCPs is addressed in the PSL PRA with respect to requiring operator action to recover the CCW flow or remove the RCPs from service. The Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function is not a significant risk contributor to core damage frequency and offsite releases in the PSL PRA. Therefore, the Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function does not

represent a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the selection criteria have not been satisfied for Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function, the requirements of this reactor trip instrument may be relocated to a licensee controlled document outside the Technical Specifications. Other reactor trip functions that are necessary to protect against violating the core specified acceptable fuel design limits or breaching the RCPB during AOOs are retained in the Technical Specifications (e.g., Reactor Coolant Flow – Low reactor trip and manual reactor trip functions).

Current Technical Specification (CTS) 3/4.1.1.3, Boron Dilution

DISCUSSION:

CTS 3/4.1.1.3 establishes a minimum flow rate to the reactor pressure vessel (RPV) of at least 3000 gpm whenever a reduction in Reactor Coolant System (RCS) boron concentration is being made. This minimum flowrate provides adequate mixing, prevents stratification, and ensures that reactivity changes will be gradual during boron concentration changes in the RCS.

A minimum boron dilution flow rate to the RPV is not assumed as an initial condition of a design basis accident (DBA) or transient analysis nor is it an assumed value to mitigate a DBA or transient (e.g., a boron dilution event).

COMPARISON TO SELECTION CRITERIA:

1. The minimum boron dilution flow rate to the RPV is not an instrumentation system. Therefore, this Specification does not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The minimum boron dilution flow rate to the RPV is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. This Specification specifies a minimum flowrate value that provides adequate mixing, prevents stratification, and ensures that reactivity changes in the core are gradual during boron concentration changes in the RCS. These limits, however, do not reflect initial condition assumptions in a DBA or transient.

3. The minimum boron dilution flow rate to the RPV is a parameter and not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The minimum boron dilution flow rate to the RPV is a parameter requirement and is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the selection criteria have not been satisfied, the Boron Dilution Specification may be relocated to a licensee controlled document outside the Technical Specifications. Control element assembly insertion limits during power operation, shutdown margin requirements during shutdown, and boration concentration requirements during refueling are retained in separate Technical Specifications and ensure adequate excess negative core reactivity is available in the event of an inadvertent boron dilution event. Additionally, RCS circulation requirements to provide mixing and prevent stratification are retained in separate Technical Specifications (i.e., shutdown cooling specifications).

CTS 3/4.1.2.1, Flow Paths – Shutdown; CTS 3/4.1.2.2, Flow Paths – Operating;
CTS 3/4.1.2.3, Charging Pumps – Shutdown; CTS 3/4.1.2.4, Charging Pumps – Operating;
CTS 3/4.1.2.5, Boric Acid Pumps – Shutdown; CTS 3/4.1.2.6, Boric Acid Pumps – Operating;
CTS 3/4.1.2.7, Borated Water Sources – Shutdown; CTS 3/4.1.2.8, Borated Water Sources – Operating

DISCUSSION:

The components associated with the boration system Technical Specifications provide the means to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin during normal operations. To accomplish this functional requirement, the current boration system technical specifications require a source of borated water, one or more flow paths to inject borated water into the RCS, and either a charging pump or high pressure safety injection pump to provide the necessary charging head.

The RCS boration management system functions to control boron concentration and maintain shutdown margin are not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of a component in the boration systems which causes a boron dilution event, response by the operator is to close the appropriate valves in the reactor makeup system. The plant response to a boron dilution event also includes control rod assembly movement and reactor trip features to ensure shutdown margin is maintained. The boration capabilities of the boration systems are not assumed to mitigate the boron dilution event.

COMPARISON TO SELECTION CRITERIA:

1. The boration systems do not constitute instrumentation systems that are used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The boration systems are not process variables, design features, or operating restrictions that represent an initial condition of a DBA or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

3. The RCS boration management system functions to control boron concentration and maintain shutdown margin do not represent a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The PSL at-power PRA has shown that the RCS boration management system functions are not significant risk contributors to core damage frequency and offsite releases. PSL does not have shutdown PRA model. However, operational experience has shown that the boration management system is not a constraint of prime importance in the mitigation of any accident or transient that results in challenging public health and safety. Therefore, the RCS boration management system functions to control boron concentration and maintain shutdown margin do not represent structures, systems, or components which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the selection criteria have not been satisfied for the RCS boration management system functions to control boron concentration and maintain shutdown margin, the Boration System Specifications may be relocated to a licensee controlled document outside the Technical Specifications. RCS boration structures, systems, or components credited as the primary success path which function or actuate to mitigate a DBA or transient are retained in separate Technical Specifications (e.g., Emergency Core Cooling System). Additionally, shutdown margin requirements during shutdown, and boration concentration requirements during refueling are retained in separate Technical Specifications to ensure adequate excess negative core reactivity is available in the event of an inadvertent boron dilution event.

Unit 2 CTS 3/4.1.3.3, Position Indication Channels – Shutdown

DISCUSSION:

When the reactor is shutdown, one CEA position indication channel for each withdrawn CEA must be OPERABLE when the reactor trip breakers are closed and one or more CEAs are withdrawn. The purpose of this requirement is to provide the control room operator with indication of the position of a CEA when the CEA is not fully inserted and perform any related operations that are required. The CEA position indication channels do not provide any automatic function and no operator action assumed in accident or transient analyses (e.g., uncontrolled CEA withdrawal event) is initiated based on CEA position. Shutdown margin requirements ensure adequate excess negative reactivity is available to maintain the reactor subcritical when a CEA with the highest reactivity worth is withdrawn while the reactor is shutdown.

COMPARISON TO SELECTION CRITERIA:

1. The CEA position monitoring system when the reactor is shutdown is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The CEA position monitoring system when the reactor is shutdown is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

3. The CEA position monitoring system when the reactor is shutdown is not a structure, system, or component that is part of the primary success path or which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The CEA position monitoring system when the reactor is shutdown and the reactor trip breakers are closed is not addressed in the PSL PRA and does not represent a system which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the selection criteria have not been satisfied for the CEA position monitoring system when the reactor trip breakers are closed and one or more CEAs are withdrawn, the requirements of the CEA Position Indication – Shutdown Specification may be relocated to a licensee controlled document outside the Unit 2 Technical Specifications. Shutdown Margin requirements during shutdown are retained in separate Technical Specifications and ensure adequate excess negative core reactivity is available in the event of a CEA system malfunction during shutdown (e.g., uncontrolled CEA withdrawal event from a subcritical condition).

Unit 1 CTS Table 3.3-11, Accident Monitoring Instrumentation, Instruments 2 through 6, and 8
Unit 2 CTS Table 3.3-10, Accident Monitoring Instrumentation, Instruments 6, 7, and 9 through 15

DISCUSSION:

Accident monitoring instrumentation ensures sufficient information is available following an accident to allow an operator to verify the response of automatic safety systems, and to take preplanned manual actions to accomplish a safe shutdown of the plant. The NRC position on application of the screening criteria to post-accident monitoring instrumentation is documented in a letter dated May 9, 1988 from T.E. Murley (NRC) to the Owners Groups (Reference 2). Regulatory Guide (RG) 1.97, Type A variables provide primary information; i.e., information that is essential for the direct accomplishment of the specified manual actions (including long term recovery actions) for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs or transients. Additionally, it could not be confirmed that RG 1.97, non-Type A Category 1 variables are not of prime importance in limiting risk. Therefore, the NRC position is that the post-accident monitoring instrumentation list should contain Type A instruments and non-Type A Category 1 instruments specified in the plant's safety evaluation report (SER) on RG 1.97. Accordingly, this position has been applied to the PSL Units 1 and 2 RG 1.97 instruments. R1

A review of the PSL Units 1 and 2 UFSAR and the NRC Regulatory Guide 1.97 Safety Evaluation for PSL Units 1 and 2 shows that the following Unit 1 CTS Table 3.3-11 and Unit 2 CTS Table 3.3.10 Instruments do not meet Category 1 or Type A requirements:

Unit 1 CTS Table 3.3-11

Instrument 2	Auxiliary Feedwater Flow Rate
Instrument 3	RCS Subcooling Margin Monitor
Instrument 4	PORV Position Indicator – Acoustic Flow Monitor
Instrument 5	PORV Block Valve Position Indicator
Instrument 6	Safety Valve Position Indicator
Instrument 8	Containment Sump Level

Unit 2 CTS Table 3.3-10

Instrument 6	Steam Generator Pressure
Instrument 7	Steam Generator Water Level – Narrow Range
Instrument 9	Refueling Water Tank Water Level
Instrument 10	Auxiliary Feedwater Flow Rate
Instrument 11	RCS Subcooling Margin Monitor
Instrument 12	PORV Position/Flow Indicator
Instrument 13	PORV Block Valve Position Indicator
Instrument 14	Safety Valve Position/Flow Indicator
Instrument 15	Containment Sump Water Level (Narrow Range)

COMPARISON TO SELECTION CRITERIA:

1. These instruments are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary.

2. The monitored parameters are not process variables, design features, or operating restrictions that are initial conditions of a DBA or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

3. These instruments are not structures, systems, or components that are part of the primary success path or which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. These post-accident monitoring instrument functions are not addressed in the PSL PRA and do not represent a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the selection criteria have not been satisfied for instruments which do not meet RG 1.97 Type A or non-Type A, Category 1 variable requirements, the requirements of these instruments may be relocated to a licensee controlled document outside the Technical Specifications. Post-accident monitoring instruments that monitor RG 1.97 Type A variables and non-Type A, Category 1 variables are retained in the Technical Specifications.

Unit 1 CTS 3/4.4.9.2, Pressurizer
Unit 2 CTS 3/4.4.9.2, Pressurizer Heatup and Cooldown Limits

DISCUSSION:

The limitation on the pressurizer pressure and temperature (P/T) ensures that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements. As such, the Technical Specification places limits on variables consistent with structural analysis results. These limits do not represent an initial condition assumption of a DBA or transient. Although the limits represent operating restrictions, and Criterion 2 includes operating restrictions, the Criterion 2 discussion of the Final Policy Statement specified that only those operating restrictions required to preclude unanalyzed accidents and transients be included in Technical Specifications.

COMPARISON TO SELECTION CRITERIA:

1. Pressurizer P/T limits are ASME Code parameter limits and are not instrumentation used for, or capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary.

2. Pressurizer P/T limits are ASME Code parameter limits and do not represent a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Pressurizer P/T limits are ASME Code parameter limits and do not represent a structure, system or component that is part of a primary success path in the mitigation of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Pressurizer P/T limits are ASME Code parameter limits and do not represent a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the selection criteria have not been satisfied, the Unit 1 Pressurizer Specification, which specifies pressurizer heatup and cooldown limits, and the Unit 2 Pressurizer Heatup and Cooldown Limits Specification may be relocated to a licensee controlled document outside the Technical Specifications. Pressure and temperature limits associated with the limiting RCS pressure boundary components are retained in separate Technical Specifications. ASME code requirements associated with the limits on pressurizer heatup and cooldown will continue to be controlled pursuant 10 CFR 50.55a.

Unit 1 CTS 3/4.4.15, Reactor Coolant System Vents
Unit 2 CTS 3/4.4.10, Reactor Coolant System Vents

DISCUSSION:

Reactor Coolant System (RCS) vents are provided to exhaust non-condensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The function, capabilities, and testing requirements of the RCS vent system are consistent with the requirements of Item II.b.1 of NUREG-0737, “Clarification of TMI [Three Mile island] Action Plan Requirements,” November 1980.

The pressurizer safety valves provide primary relief capability to exhaust non-condensible gases and steam from the RCS during a DBA or transient at power. The pressurizer power operated relief valves (PORVs) and other vent paths (e.g., pressurizer manway cover removed) provide vent capability during RCS low temperature and pressure conditions.

COMPARISON TO SCREENING CRITERIA:

1. The RCS vents do not function as installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The RCS vents are not process variables, design features, or operating restrictions that represents an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The RCS vents are not a structures, systems or components that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The RCS vent function to preclude the inhibition of natural circulation core cooling is not a significant risk contributor to core damage frequency and offsite releases and this TMI function does not represent a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied for the RCS vent function that precludes the inhibition of natural circulation core cooling, the Reactor Coolant System Vents Specification may be relocated to a licensee controlled document outside Technical Specifications. Pressure relief requirements during power operation and during low temperature and pressure conditions are retained in separate Technical Specifications and ensure relief capability is provided to exhaust non-condensible gases and steam from the primary system. In addition, this NUREG-0737 requirement is required pursuant to 10 CFR 50.34(f)(2)(vi). Compliance with applicable portions of 10 CFR 50.34(f)(2) is required by the operating licenses of PSL Units 1 and 2.

CTS 3/4.7.2, Steam Generator Pressure/Temperature Limitation

DISCUSSION:

The limitation on steam generator pressure and temperature (P/T) ensures that pressure-induced stresses on the steam generators do not exceed the maximum allowable fracture toughness limits. These pressure and temperature limits are based on maintaining a steam generator reference temperature for Nil Ductility Transition (RT_{NDT}) sufficient to prevent brittle fracture. As such, the Technical Specification places limits on variables consistent with structural analysis results. These limits do not represent an initial condition assumption of a DBA or transient. Although the limits represent operating restrictions, and Criterion 2 includes operating restrictions, the Criterion 2 discussion of the Final Policy Statement specified that only those operating restrictions required to preclude unanalyzed accidents and transients be included in Technical Specifications.

COMPARISON TO SELECTION CRITERIA:

1. Steam generator P/T limits are ASME Code parameter limits and are not instrumentation used for, or capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary.

2. Steam generator P/T limits are ASME Code parameter limits and do not represent a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Steam generator P/T limits are ASME Code parameter limits and not a structure, system, or component that is part of a primary success path in the mitigation of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Steam generator P/T limits are ASME Code parameter limits and are not structures, systems, or components which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the selection criteria have not been satisfied, the Steam Generator Pressure/Temperature Limitation Specification may be relocated to a licensee controlled document outside the Technical Specifications. Pressure and temperature limits associated with the limiting RCS pressure boundary components are retained in separate Technical Specifications. ASME code requirements associated with the limits on steam generator heatup and cooldown will continue to be controlled pursuant 10 CFR 50.55a

Unit 2 CTS 3/4.7.6, Flood Protection

DISCUSSION:

External flooding during hurricane conditions is not designated as a PSL DBA or transient event. In addition, external flooding due to a hurricane is not postulated to occur during any DBA or transient, thus water level (as it pertains to wave run-up effects during a hurricane) is not credited in any safety analysis. The Flood Protection Technical Specification requirements ensure that facility protective actions will be taken in the event of flood conditions whenever a hurricane warning is issued. The installation of the stoplogs ensures adequate protection for wave run-up effects where no permanent adjacent structures exist and provides protection to safety-related equipment.

COMPARISON TO SCREENING CRITERIA:

1. Flood protection requirements are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. Flood protection requirements are not process variables that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Flood protection requirements are not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The specific requirements specified in the Flood Protection Specification related to a hurricane are found to be non-significant risk contributors to core damage frequency and offsite releases and do not represent structures, systems, or components which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the Flood Protection Specification may be relocated to a licensee controlled document outside the Unit 2 Technical Specifications. A general requirement associated with hazard barriers (ITS LCO 3.0.9), which includes external flooding barriers, is added to ITS Section 3.0. A hazard barrier that cannot perform its related support function will be evaluated and managed under the Maintenance Rule plant configuration control requirement, 10 CFR 50.65(a)(4), and associated industry guidance, NUMARC 93-01, Revision 4A.

Unit 1 CTS 3/4.7.9, Sealed Source Contamination
Unit 2 CTS 3/4.7.10, Sealed Source Contamination

DISCUSSION:

The limitations on sealed source contamination are intended to ensure that the total body and individual organ irradiation doses do not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of < 0.005 microcuries of removable contamination on each sealed source. This requirement and the associated surveillance requirements bear no relation to the conditions or limitations that are necessary to ensure safe reactor operation.

COMPARISON TO SELECTION CRITERIA:

1. The sealed source contamination limitation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary.

2. The sealed source contamination limitation is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The sealed source contamination limitation is not part of a primary success path in the mitigation of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The sealed source contamination limitation is not discussed in the PSL PRA and is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the selection criteria have not been satisfied, the Sealed Source Contamination Specification may be relocated to a licensee controlled document outside the Technical Specifications. Requirements associated with the sealed sources are governed by 10 CFR Part 70. Compliance with applicable portions of 10 CFR Part 70 is required by the operating licenses of PSL Units 1 and 2.

ENCLOSURE 2

VOLUME 2

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

GENERIC DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION AND ENVIRONMENTAL ASSESSMENT

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
ADMINISTRATIVE CHANGES

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants" Some of the proposed changes involve reformatting, renumbering, and rewording of current technical specifications (CTS) with no change in intent. These changes, since they do not involve technical changes to the CTS, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording that does not affect the technical content of the CTS. These changes also include non-technical modifications of requirements to conform to TSTF-GG-05-01, "Writer's Guide for Plant-Specific Improved Standard Technical Specifications," or provide consistency with the Improved Standard Technical Specifications in NUREG-1432. Administrative changes are not intended to add, delete, or relocate any technical requirements of the CTS.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves reformatting, renumbering, and rewording the CTS. The reformatting, renumbering, and rewording process involves no technical changes to the CTS. As such, this change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accident or transient events.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. This change is administrative in nature.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
MORE RESTRICTIVE CHANGES

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve adding more restrictive requirements to the Current Technical Specifications by either making current requirements more stringent or by adding new requirements that currently do not exist.

These changes include additional requirements that decrease allowed outage times, increase the Frequency of Surveillances, impose additional Surveillances, increase the scope of Specifications to include additional plant equipment, increase the Applicability of Specifications, or provide additional Actions. These changes are made to conform to NUREG-1432 and have been evaluated to not be detrimental to plant safety.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change provides more stringent Technical Specification requirements for the facility. These more stringent requirements do not result in operations that significantly increase the probability of initiating an analyzed event, and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change does impose different Technical Specification requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The imposition of more restrictive requirements either has no effect on or increases the margin of plant safety. As provided in the discussion of change, each change in this category is, by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
RELOCATED SPECIFICATIONS

The Florida Power & Light Company (FPL) is converting St. Lucie Plant (PSL) to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve relocating Current Technical Specification (CTS) Limiting Conditions for Operations (LCOs) to licensee controlled documents.

FPL has evaluated the CTS using the criteria set forth in 10 CFR 50.36. Specifications identified by this evaluation that did not meet the retention requirements specified in the regulation are not included in the ITS. These specifications have been relocated from the CTS to a licensee controlled document (e.g., Technical Requirements Manual (TRM) or Updated Final Safety Analysis Report (UFSAR)).

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relocates requirements and Surveillances for structures, systems, components, or variables that do not meet the criteria of 10 CFR 50.36(c)(2)(ii) for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the PSL Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and Surveillances for these affected structures, systems, components, or variables will be relocated from the CTS to a licensee controlled document that is maintained pursuant to 10 CFR 50.59. The UFSAR is subject to the change control provisions of 10 CFR 50.59 and 10 CFR 50.71(e). In addition, the affected structures, systems, components, or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59, and are subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change in the methods governing

normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of existing requirements will be maintained.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change will not reduce a margin of safety because it has no significant effect on any safety analyses assumptions, as indicated by the fact that the requirements do not meet the 10 CFR 50.36 criteria for retention. In addition, the relocated requirements are moved without change, and any future changes to these requirements will be evaluated per 10 CFR 50.59.

NRC prior review and approval of changes to these relocated requirements, in accordance with 10 CFR 50.92, will no longer be required. This review and approval does not provide a specific margin of safety that can be evaluated. However, the proposed change is consistent with NUREG-1432, issued by the NRC, which allows revising the CTS to relocate these requirements and Surveillances to a licensee controlled document.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
REMOVED DETAIL CHANGES

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve moving details out of the Current Technical Specifications (CTS) and into the Technical Specifications Bases, the Updated Final Safety Analysis Report (UFSAR), the Containment Leakage Rate Testing Program, the Technical Requirements Manual (TRM), or other documents under regulatory control, such as the Offsite Dose Calculation Manual, the Quality Assurance Topical Report, the Inservice Testing Program, the Inservice Inspection Program, and the Surveillance Frequency Control Program. The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. The removal of this information conforms to NUREG-1432 for format and content.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relocates certain details from the CTS to other documents under regulatory control. The Technical Specification Bases and the TRM will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.59 and 10 CFR 50.71(e). Other documents are subject to controls imposed by the ITS or other regulations. Since any changes to these documents will be evaluated, no significant increase in the probability or consequences of an accident previously evaluated will be allowed.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. In addition, the details to be moved from the CTS to other documents are not being changed. Since any future changes to these details will be evaluated under the applicable regulatory change control mechanism, no significant reduction in a margin of safety will be allowed. A significant reduction in the margin of safety is not associated with the elimination of the 10 CFR 50.90 requirement for NRC review and approval of future changes to the relocated details. Not including these details in the Technical Specifications is consistent with NUREG-1432, issued by the NRC, which allows revising the Technical Specifications to relocate these requirements and Surveillances to a licensee controlled document controlled by 10 CFR 50.59, 10 CFR 50.71(e), or other Technical Specification controlled or regulation controlled documents.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 1
RELAXATION OF LCO REQUIREMENTS

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve relaxation of the Current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by the elimination of specific items from the LCO or Tables referenced in the LCO, or the addition of exceptions to the LCO.

These changes reflect the Standard Technical Specifications (ISTS) approach to provide LCO requirements that specify the protective conditions that are required to meet safety analysis assumptions for required features. These conditions replace the lists of specific devices used in the CTS to describe the requirements needed to meet the safety analysis assumptions. The ITS also includes LCO Notes which allow exceptions to the LCO for the performance of testing or other operational needs. The ITS provides the protection required by the safety analysis, and provides flexibility for meeting the conditions without adversely affecting operations since equivalent features are required to be OPERABLE. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. These changes are made to conform with NUREG-1432, and have been evaluated to not be detrimental to plant safety.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change provides less restrictive LCO requirements for operation of the facility. These less restrictive LCO requirements do not result in operation that will significantly increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the current safety analyses and licensing basis.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the change is consistent with the assumptions in the current safety analyses and licensing basis.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The imposition of less restrictive LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the current safety analyses and licensing basis requirements are maintained.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 2
RELAXATION OF APPLICABILITY

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve relaxation of the applicability of Current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by reducing the conditions under which the LCO requirements must be met.

CTS requirements are being eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements that are indeterminate or which are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the ITS may be satisfied by exiting the applicability which takes the plant out of the conditions that require the safety system to be OPERABLE.

This change provides the protection required by the safety analyses, and provides flexibility for meeting limits by restricting the application of the limits to the conditions assumed in the safety analyses. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. The change is made to conform with NUREG-1432, and has been evaluated to not be detrimental to plant safety.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes the conditions under which the LCO requirements for operation of the facility must be met. These less restrictive applicability requirements for the LCOs do not result in operation that will significantly increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure that process variables, structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the requirements are consistent with the assumptions in the safety analyses and licensing basis.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The relaxed applicability of LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the LCO requirements are applied in the MODES and specified conditions assumed in the safety analyses and licensing basis.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 3
RELAXATION OF COMPLETION TIME

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve relaxation of the Completion Times for Required Actions in the Current Technical Specifications.

Upon discovery of a failure to meet a Limiting Condition for Operation (LCO), the ITS specifies times for completing Required Actions of the associated Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken within specified Completion Times. These times define limits during which operation in a degraded condition is permitted. Adopting Completion Times from the ITS is acceptable because the Completion Times take into account the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a Design Basis Accident (DBA) occurring during the repair period. In addition, the ITS provides consistent Completion Times for similar conditions. These changes are made to conform with NUREG-1432, and have been evaluated to not be detrimental to plant safety.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes the Completion Time for a Required Action. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated, and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Completion Time does not significantly increase the probability of any accident previously evaluated. The consequences of an analyzed accident during the relaxed Completion Time are the same as the consequences during the existing Completion Time. As a result, the consequences of any accident previously evaluated are not significantly increased.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the method governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The relaxed Completion Time for a Required Action does not involve a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the allowed Completion Time is consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 4
RELAXATION OF REQUIRED ACTION

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve relaxation of the Required Actions in the Current Technical Specifications.

Upon discovery of a failure to meet a Limiting Condition for Operation (LCO), the ITS specifies Required Actions to complete for the associated Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken in response to the degraded conditions. These actions minimize the risk associated with continued operation while providing time to repair inoperable features. Some of the Required Actions are modified to place the plant in a MODE in which the LCO does not apply. Adopting Required Actions from NUREG-1432 is acceptable because the Required Actions take into account the OPERABILITY status of redundant systems of required features, the capacity and capability of the remaining features, and the compensatory attributes of the Required Actions as compared to the LCO requirements. These changes are made to conform with NUREG-1432, and have been evaluated to not be detrimental to plant safety.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes Required Actions. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated, and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Required Actions do not significantly increase the probability of any accident previously evaluated. The Required Actions in the ITS have been developed to provide appropriate remedial actions to be taken in response to the degraded condition considering the OPERABILITY status of the redundant systems of required features, and the capacity and capability of remaining features while minimizing the risk associated with continued operation. As a result, the consequences of any accident previously evaluated are not significantly increased.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The relaxed Required Actions do not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to minimize the risk of continued operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a Design Basis Accident (DBA) occurring during the repair period.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 5
DELETION OF SURVEILLANCE REQUIREMENT

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve deletion of Surveillance Requirements in the Current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified OPERABLE prior to entering applicable operating conditions. The ITS eliminates unnecessary CTS Surveillance Requirements that do not contribute to verification that the equipment used to meet the Limiting Condition for Operation (LCO) can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety functions. These changes are made to conform with NUREG-1432, and have been evaluated to not be detrimental to plant safety.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change deletes Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing the accident mitigation functions assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not significantly affected.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The remaining Surveillance Requirements are consistent with industry practice, and are considered sufficient to prevent the removal of the subject Surveillances from creating a new or different type of accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The deleted Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the deleted Surveillance Requirements are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 6
RELAXATION OF SURVEILLANCE REQUIREMENT ACCEPTANCE CRITERIA

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve the relaxation of Surveillance Requirements acceptance criteria in the Current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified OPERABLE prior to entering applicable operating conditions. The ITS eliminates or relaxes the Surveillance Requirement acceptance criteria that do not contribute to verification that the equipment used to meet the Limiting Condition for Operation (LCO) can perform its required functions. For example, the ITS allows some Surveillance Requirements to verify OPERABILITY under actual or test conditions. Adopting the ITS allowance for "actual" conditions is acceptable because required features cannot distinguish between an "actual" signal or a "test" signal. Also included are changes to CTS requirements that are replaced in the ITS with separate and distinct testing requirements that when combined, include OPERABILITY verification of all components required in the LCO for the features specified in the CTS. Adopting this format preference in the ITS is acceptable because Surveillance Requirements that remain include testing of all previous features required to be verified OPERABLE. Changes that provide exceptions to Surveillance Requirements to provide for variations that do not affect the results of the test are also included in this category. These changes are made to conform with NUREG-1432, and have been evaluated to not be detrimental to plant safety.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes the acceptance criteria of Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing the accident mitigation functions assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not significantly affected.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The relaxed acceptance criteria for Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxed Surveillance Requirement acceptance criteria have been evaluated to ensure that they are sufficient to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner that gives confidence that the equipment can perform its assumed safety function.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 7
RELAXATION OF SURVEILLANCE FREQUENCY

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve the relaxation of Surveillance Frequencies in the Current Technical Specifications (CTS).

CTS and ITS Surveillance Frequencies specify time interval requirements for performing Surveillance tests. Increasing the time interval between Surveillance tests in the ITS results in decreased equipment unavailability due to testing which also increases equipment availability. In general, the ITS contain Surveillance Frequencies that are consistent with industry practice or industry standards for achieving acceptable levels of equipment reliability. Adopting testing practices specified in the ITS is acceptable based on similar design, like-component testing for the system application and the availability of other ITS requirements which provide regular checks to ensure limits are met. Relaxation of Surveillance Frequency can also include the addition of Surveillance Notes which allow testing to be delayed until appropriate unit conditions for the test are established, or exempt testing in certain MODES or specified conditions in which the testing cannot be performed.

Reduced testing can result in a safety enhancement because the unavailability due to testing is reduced, and reliability of the affected structure, system or component should remain constant or increase. Reduced testing is acceptable where operating experience, industry practice, or the industry standards such as manufacturers' recommendations have shown that these components usually pass the Surveillance when performed at the specified interval, thus the Surveillance Frequency is acceptable from a reliability standpoint. Surveillance Frequency changes to incorporate alternate train testing have been shown to be acceptable where other qualitative or quantitative test requirements are required that are established predictors of system performance. Surveillance Frequency extensions can be based on NRC-approved topical reports. The NRC staff has accepted topical report analyses that bound the plant-specific design and component reliability assumptions. These changes are made to conform with NUREG-1432, and have been evaluated to not be detrimental to plant safety.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes Surveillance Frequencies. The relaxed Surveillance Frequencies have been established based on achieving acceptable levels of equipment reliability. Consequently, equipment that could initiate an accident previously evaluated will continue to operate as expected, and the probability of the

initiation of any accident previously evaluated will not be significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing any accident mitigation functions assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not significantly affected.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The relaxed Surveillance Frequencies do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxation in the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at a Frequency that gives confidence that the equipment can perform its assumed safety function when required.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 8
DELETION OF SURVEILLANCE REQUIREMENT SHUTDOWN PERFORMANCE REQUIREMENTS

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition in the Current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified OPERABLE periodically. The CTS requires these Surveillances to be performed with the unit in a specified condition, usually in a condition outside the Applicability of the Limiting Condition for Operation (LCO). The ITS Surveillance does not include the restriction on unit conditions. The control of the unit conditions appropriate to perform the test is an issue for procedures and scheduling, and has been determined by the NRC Staff to be unnecessary as an ITS restriction. As indicated in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991, allowing this control is consistent with the vast majority of other Technical Specification Surveillances that do not dictate unit conditions for the Surveillance. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. These changes are generally made to conform with NUREG-1432, and have been evaluated to not be detrimental to plant safety.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The appropriate plant conditions for performance of the Surveillance will continue to be controlled in plant procedures to assure the potential consequences are not significantly increased. This control method has been previously determined to be acceptable as indicated in NRC Generic Letter No. 91-04. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change involves the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition, but does not change the method of performance. The appropriate plant conditions for performance of the Surveillance will continue to be controlled in plant procedures to assure the possibility of a new or different kind of accident is not created. The control method has been previously determined to be acceptable as indicated in NRC Generic Letter No. 91-04.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change involves the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition. However, the appropriate plant conditions for performance of the Surveillance will continue to be controlled in plant procedures. The control method has been previously determined to be acceptable as indicated in NRC Generic Letter No. 91-04.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 9
DELETION OF REPORTING REQUIREMENTS

The Florida Power & Light Company (FPL) is converting St. Lucie Plant to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5.0, "Standard Technical Specifications, Combustion Engineering Plants." Some of the proposed changes involve the deletion of requirements in the Current Technical Specifications (CTS) to send reports to the NRC.

The CTS includes requirements to submit reports to the NRC under certain circumstances. However, the ITS eliminates these requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The ITS changes to reporting requirements are acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change has no effect on the safe operation of the plant. These changes are generally made to conform with NUREG-1432 and have been evaluated to not be detrimental to plant safety.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change deletes reporting requirements. Sending reports to the NRC is not an initiator of any accident previously evaluated. Consequently, the probability of any accident previously evaluated is not significantly increased. Sending reports to the NRC has no effect on the ability of equipment to mitigate an accident previously evaluated. As a result, the consequences of any accident previously evaluated is not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The deletion of reporting requirements does not result in a significant reduction in the margin of safety. The ITS eliminates the requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The change to reporting requirements does not affect the margin of safety because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

ENVIRONMENTAL ASSESSMENT

The Florida Power & Light Company (FPL) has evaluated this license amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. FPL has determined that this license amendment meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in the generic and specific Determination of No Significant Hazards Considerations, this proposed amendment does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed amendment does not affect the generation of any radioactive effluents, and does not affect any of the permitted effluent release paths.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not cause radiological exposure in excess of the dose criteria for restricted and unrestricted access specified in 10 CFR 20. Individual worker exposures will be maintained within acceptable limits by the St. Lucie Plant Radiation Protection Program.

Therefore, pursuant to 10 CFR 51.22 (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

ENCLOSURE 2

VOLUME 3

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 1.0
USE AND APPLICATION

Revision 1

LIST OF ATTACHMENTS

1. ITS Chapter 1.0, Definitions

ATTACHMENT 1

ITS Chapter 1.0, Definitions

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

1.0 DEFINITIONS

NOTE The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

A12

ACTIONS

ACTION

1.1 ACTION shall be that part of a specification which prescribes Required Actions to be taken that are remedial measures required under designated conditions within specified Completion Times.

A01

AXIAL SHAPE INDEX (ASI)

AXIAL SHAPE INDEX (ASI)

1.2 The AXIAL SHAPE INDEX (Y_E) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

LA01

$$Y_E = \frac{L - U}{L + U} \qquad Y_I = AY_E + B$$

AZIMUTHAL POWER TILT – T_q

AZIMUTHAL POWER TILT – T_q

1.3 AZIMUTHAL POWER TILT shall be the maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

$$\text{AZIMUTHAL POWER TILT} = \max\left[\frac{\text{Power in any core quadrant (upper or lower)}}{\text{Average power of all quadrants (upper or lower)}}\right] - 1$$

A01

CHANNEL CALIBRATION

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

A02

CHANNEL CHECK

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

A02

A02 **INSERT 1**

Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

A02 **INSERT 2**

, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

1.1 → **DEFINITIONS**

CHANNEL FUNCTIONAL TEST

CHANNEL FUNCTIONAL TEST

L01

or actual

~~1.6~~ A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the ~~primary~~ sensor as practicable to verify OPERABILITY ~~including alarm and/or trip functions.~~

A01

Insert 3

A03

CONTAINMENT VESSEL INTEGRITY

A04

1.7 CONTAINMENT VESSEL INTEGRITY shall exist when:

See ITS 3.6.1

a. All containment vessel penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed position except for valves that are open on an intermittent basis under administrative control.

See ITS 3.6.3

b. All containment vessel equipment hatches are closed and sealed,

c. Each containment vessel air lock is in compliance with the requirements of Specification 3.6.1.3,

See ITS 3.6.2

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

~~CONTROLLED LEAKAGE~~

~~1.8 CONTROLLED LEAKAGE shall be the seal water flow supplied from the reactor coolant pump seals.~~

~~CORE ALTERATION~~

~~1.9 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include shared (4 fingered) control element assemblies (CEAs) withdrawn into the upper guide structure (UGS) or evolutions performed with the UGS in place such as CEA latching/unlatching or verification of latching/unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~

A04

CORE OPERATING LIMITS REPORT (COLR)

CORE OPERATING LIMITS REPORT (COLR)

~~1.9a~~ THE COLR is the unit-specific document that provides cycle specific parameter limits for the current ~~operating~~ reload cycle. These cycle-specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.3 → ~~6.9.1.11~~. Plant operation within these limits is addressed in individual Specifications.

A01

A03

INSERT 3

of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

1.1 DEFINITIONS

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131

when inhaled as the combined activities of iodine isotopes

microcuries per gram

that

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (~~μCi/gram~~) ~~which~~ alone would produce the same ~~thyroid~~ dose ~~as the quantity and isotopic mixture of~~ I-131, I-132, I-133, I-134 and I-135 actually present. The ~~thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."~~

A01

determination of DOSE EQUIVALENT I-131 shall be performed using Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11.

DOSE EQUIVALENT XE-133

DOSE EQUIVALENT XE-133

microcuries per gram

1.11 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (~~μCi/gram~~) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, “External Exposure to Radionuclides in Air, Water, and Soil.”

ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME

(ESF)

ENGINEERED SAFETY FEATURES RESPONSE TIME

A01

ESF

S

1.12 The ~~ENGINEERED SAFETY FEATURES~~ REPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC~~.~~

,

, or the components have been evaluated in accordance with an NRC approved methodology.

L02

~~FREQUENCY NOTATION~~

LA02

~~1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.~~

~~GASEOUS RADWASTE TREATMENT SYSTEM~~

~~1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~

A04

1.1 **DEFINITIONS**

LEAKAGE

~~IDENTIFED~~ LEAKAGE

~~1.15~~ ~~IDENTIFIED~~ LEAKAGE shall be:

a. Identified LEAKAGE — is

1. ~~a.~~ ~~Leakage (except CONTROLLED~~ LEAKAGE~~) into closed systems~~, such as pump seal or valve packing ~~leaks~~ that ~~are~~ captured~~,~~ and conducted to a sump or collecting tank~~,~~ ~~or~~

(except reactor coolant pump (RCP) seal water injection or leakoff),

that from — s

collection systems or

;

A01

2. ~~b.~~ Leakage into the containment atmosphere from sources that are both specifically located and known ~~either not~~ to interfere with the operation of leakage detection systems ~~or not to be PRESSURE BOUNDARY LEAKAGE,~~ or

A05

(RCS) — not

;

3. ~~c.~~ Reactor Coolant System leakage through a steam generator to the secondary system (Primary~~-~~to~~-~~secondary leakage)~~.~~

;

b. Unidentified LEAKAGE — see CTS 1.34 Markup

c. Pressure Boundary LEAKAGE — see CTS 1.22 Markup

INSERVICE TESTING PROGRAM

INSERVICE TESTING PROGRAM

~~1.16~~ The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

A01

~~MEMBER(S) OF THE PUBLIC~~

~~1.17~~ ~~MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose~~.

A04

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 THE OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

A04

See ITS 5.5.1

1.1 DEFINITIONS

OPERABLE – OPERABILITY

~~1.19~~ A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s)~~,~~ and when all necessary attendant instrumentation, controls, electrical power, cooling ~~or~~ seal water, lubrication ~~or~~ other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

~~OPERATIONAL~~ MODE

~~1.20~~ A~~n OPERATIONAL~~ MODE ~~(i.e., MODE)~~ shall correspond to any one inclusive combination of core reactivity condition, power level~~, and~~ average reactor coolant temperature specified in Table ~~1.2~~.

PHYSICS TESTS

~~1.21~~ PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation ~~and~~ ~~(1)~~ described in Chapter 14.0 of the FSAR, ~~(2)~~ authorized under the provisions of 10 CFR 50.59, or ~~(3)~~ otherwise approved by the Commission.

~~PRESSURE BOUNDARY~~ LEAKAGE

~~1.22~~ PRESSURE BOUNDARY LEAKAGE ~~shall be leakage~~ (except primary~~-~~to~~-~~secondary leakage) through a ~~non-isolable~~ fault in ~~a Reactor Coolant System~~ component body, pipe wall or vessel wall.

~~PROCESS CONTROL PROGRAM (PCP)~~

~~1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packing of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~

~~PURGE – PURGING~~

~~1.24 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement~~.

1.1 **DEFINITIONS**

RATED THERMAL POWER

RATED THERMAL POWER (RTP)

RTP

~~1.25 RATED THERMAL POWER~~ shall be a total reactor core heat transfer rate to the reactor coolant of 3020 MWt.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

PROTECTION (RPS)

REACTOR ~~TRIP~~ SYSTEM RESPONSE TIME

RPS s RPS that

~~1.26~~ The ~~REACTOR TRIP SYSTEM~~ RESPONSE TIME shall be ~~the~~ time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power to the CEA drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

A01

, or the components have been evaluated in accordance with an NRC approved methodology.

L02

~~REPORTABLE EVENT~~

~~1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.~~

A04

SHIELD BUILDING INTEGRITY

A04

1.28 SHIELD BUILDING INTEGRITY shall exist when:

See ITS 3.6.7

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

See ITS 3.6.9

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

SHUTDOWN MARGIN SDM

SDM

~~1.29 SHUTDOWN MARGIN~~ shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all ~~full-length control element assemblies~~ (shutdown and regulating) are fully inserted except for the single ~~assembly~~ of highest reactivity worth which is assumed to be fully withdrawn.

A01

CEA CEAs ,

See CTS 4.1.1.1.1.a Markup

A09

See CTS 4.1.1.1.1.e *Note Markup

A13

~~SITE BOUNDARY~~

~~1.30 SITE BOUNDARY means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.~~

A04

~~SOURCE CHECK~~

~~1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.~~

1.1 DEFINITIONS

~~**STAGGERED TEST BASIS**~~

~~1.32 A STAGGERED TEST BASIS shall consist of:~~

~~a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and~~

~~b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.~~

A04

THERMAL POWER

THERMAL POWER

~~1.33~~ THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

A01

LEAKAGE

~~UNIDENTIFIED~~ LEAKAGE

b.

~~1.34~~ UNIDENTIFIED LEAKAGE ~~shall be~~ all leakage ~~which~~ is not IDENTIFIED LEAKAGE ~~or CONTROLLED LEAKAGE.~~

(except RCP seal water injection or leakoff) that

; and

A01
A05
A04

~~**UNRESTRICTED AREA**~~

~~1.35 UNRESTRICTED AREA means an area, access to which is neither limited nor controlled by the licensee.~~

~~**UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F$_r$**~~

~~1.36 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.~~

A04

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours
D	At least once per 24 hours
W	At least once per 7 days
4/M*	At least 4 per month at intervals of no greater than 9 days and a minimum of 48 per year
M	At least once per 31 days
Q	At least once per 92 days
SA	At least once per 184 days
R	At least once per 18 months
S/U	Prior to each reactor startup
P**	Completed prior to each release
SFCP	In accordance with the Surveillance Frequency Control Program
N.A.	Not applicable

LA02

* For Radioactive Effluent Sampling

** For Radioactive Batch Releases Only

TABLE ~~1.2~~ 1.1-1 (page 1 of 1)

~~OPERATIONAL~~ MODES

MODE (TITLE)	REACTIVITY CONDITION (~~,~~ ( ) Keff)	% RATED THERMAL POWER (a) ~~*~~	AVERAGE COOLANT TEMPERATURE (REACTOR) (°F)
1. POWER OPERATION	≥ 0.99	> 5~~%~~	NA ~~≥ 325°F~~
2. STARTUP	≥ 0.99	≤ 5~~%~~	NA ~~≥ 325°F~~
3. HOT STANDBY	< 0.99	NA ~~0~~	≥ 325~~°F~~
4. HOT SHUTDOWN (b)	< 0.99	NA ~~0~~	325~~°F~~ > Tavg > 200~~°F~~
5. COLD SHUTDOWN (b)	< 0.99	NA ~~0~~	≤ 200°~~F~~
6. REFUELING~~**~~ (c)	NA ~~≤ 0.95~~	NA ~~0~~	NA ~~≤ 140°F~~

A01 A11 LA03

(a) ~~*~~ Excluding decay heat.

(c) ~~**~~ ~~Fuel in the reactor vessel with the~~ One or more reactor vessel head closure bolts less than fully tensioned ~~or with the head removed~~.

(b) All reactor vessel head closure bolts fully tensioned.

A01 A06

Add proposed ITS Sections
1.2 Logical Connectors
1.3 Completion Times
1.4 Frequency

A10

ITS

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

See ITS 3.1.1

SHUTDOWN MARGIN - T_{avg} > 200 °F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN not within limits immediately initiate and continue boration at ≥ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. ~~If the inoperable~~ CEA ~~is~~ not fully inserted, ~~and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable,~~ the ~~above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn~~ worth of ~~the immovable or untrippable~~ CEA~~(s)~~.

With any | s | capable of being | SHUTDOWN MARGIN | A01 | these | must be accounted for in the determination of SDM | reactivity

b. When in MODES 1 or 2#, in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2## at least once during CEA withdrawal and in accordance with the Surveillance Frequency Control Program until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

---

* See Special Test Exception 3.10.1.

\# With K_{eff} ≥ 1.0.

\## With K_{eff} < 1.0.

ITS

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, in accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,*
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

See ITS 3.1.1

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1000 pcm in accordance with the Surveillance Frequency Control Program. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

See ITS 3.1.2

SHUTDOWN MARGIN

* ~~For Modes 3 and 4, during calculation of shutdown margin~~ with all CEA~~'~~s verified fully inserted~~, the single CEA with the highest reactivity worth need not be assumed to be stuck in the fully withdrawn position~~.

L03

A13

by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation

However,

ITS

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} ≤ 200 °F

See ITS 3.1.1

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

Within the limits specified in the COLR, and in addition with the Reactor Coolant System drained below the hot leg centerline, one charging pump shall be rendered inoperable.*

APPLICABILITY: MODE 5.

ACTION:

If the SHUTDOWN MARGIN requirements cannot be met, immediately initiate and continue boration at ≥ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN requirements of Specification 3.1.1.2 shall be determined:

SHUTDOWN MARGIN

With any | s not capable of being fully inserted

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. ~~If the inoperable~~ CEA ~~is immovable or untrippable~~, the ~~above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn~~ worth of ~~the immovable or untrippable~~ CEA~~(s)~~. (A01)

reactivity

these | must be accounted for in the determination of SDM

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2.b and by verifying at least one charging pump is rendered inoperable.*

* Breaker racked-out.

1.0 DEFINITIONS

NOTE The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

Term | Definition | and Bases

A12

ACTIONS

ACTION

S

1.1 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

that | Required Actions to be taken | within specified Completion Times

A01

AXIAL SHAPE INDEX (ASI)

AXIAL SHAPE INDEX (ASI)

(ASI)

1.2 The AXIAL SHAPE INDEX (Y_E) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

ASI shall be | generated in | generated in | half of the core | half of the core, | generated in the lower and upper halves of the core.

LA01

$$Y_E = \frac{L - U}{L + U} \qquad Y_I = AY_E + B$$

ASI = (LOWER – UPPER) / LOWER + UPPER)

AZIMUTHAL POWER TILT (T_q)

AZIMUTHAL POWER TILT – T_q

1.3 AZIMUTHAL POWER TILT shall be the maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

(P_{quad}) | of the | (P_{avg}) | ,

$$\text{AZIMUTHAL POWER TILT} = \text{MAX}\left[\frac{\text{Power in any core quadrant (upper or lower)}}{\text{Average power of all quadrants (upper or lower)}} - 1\right]$$

$T_q = \text{MAX} \mid (P_{quad} - P_{avg}) / P_{avg} \mid$

A01

CHANNEL CALIBRATION

CHANNEL CALIBRATION

that

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

INSERT 1

A02

all devices in | means of | required for channel OPERABILITY

CHANNEL CHECK

CHANNEL CHECK

INSERT 2

A02

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

, by observation,

to

A02 **INSERT 1**

Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

A02 **INSERT 2**

, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

1.1 DEFINITIONS

CHANNEL FUNCTIONAL TEST

CHANNEL FUNCTIONAL TEST

~~1.6~~ A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the ~~primary~~ sensor as practicable to verify OPERABILITY ~~including alarm and/or trip functions.~~

Insert 3

L01 A01 A03

CONTAINMENT VESSEL INTEGRITY

A04

1.7 CONTAINMENT VESSEL INTEGRITY shall exist when:

See ITS 3.6.1

a. All containment vessel penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open on an intermittent basis under administrative control.

See ITS 3.6.3

b. All containment vessel equipment hatches are closed and sealed,

c. Each containment vessel air lock is in compliance with the requirements of Specification 3.6.1.3,

See ITS 3.6.2

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

~~CONTROLLED LEAKAGE~~

~~1.8 CONTROLLED LEAKAGE shall be the seal water flow supplied from the reactor coolant pump seals.~~

~~CORE ALTERATION~~

~~1.9 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include evolutions performed with the upper guide structure (UGS) in place such as control element assembly (CEA) latching/unlatching or verification of latching/unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~

A04

CORE OPERATING LIMITS REPORT (COLR)

CORE OPERATING LIMITS REPORT (COLR)

~~1.9a~~ THE COLR is the unit-specific document that provides cycle specific parameter limits for the current ~~operating~~ reload cycle. These cycle-specific parameter limits shall be determined for each reload cycle in accordance with Specification ~~6.9.1.11~~. Plant operation within these limits is addressed in individual Specifications.

5.6.3

A01

A03

INSERT 3

of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

1.1 **DEFINITIONS**

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131

~~1.10~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) ~~which~~ alone would produce the same ~~thyroid~~ dose ~~as the quantity and isotopic mixture of~~ I-131, I-132, I-133, I-134 and I-135 actually present. The ~~thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."~~

when inhaled as the combined activities of iodine isotopes

that

determination of DOSE EQUIVALENT I-131 shall be performed using Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11.

A01

DOSE EQUIVALENT XE-133

DOSE EQUIVALENT XE-133

~~1.11~~ DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (~~µ~~Ci/gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

microcuries per gram

ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME

ENGINEERED SAFETY FEATURE~~S~~ RESPONSE TIME

(ESF)

ESF

~~1.12~~ The ~~ENGINEERED SAFETY FEATURES~~ RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC~~.~~

,

, or the components have been evaluated in accordance with an NRC approved methodology.

A01

L02

~~**FREQUENCY NOTATION**~~

~~1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.~~

LA02

~~**GASEOUS RADWASTE TREATMENT SYSTEM**~~

~~1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~

A04

LEAKAGE

~~**IDENTIFIED** **LEAKAGE**~~

a. Identified LEAKAGE

~~1.15 IDENTIFIED~~ LEAKAGE shall be:

1. ~~a.~~ ~~Leakage (except CONTROLLED~~ LEAKAGE~~) into closed systems~~, such as pump seal or valve packing ~~leaks~~ that ~~are~~ captured~~,~~ and conducted to a sump or collecting tank~~, or~~

s

is

that from

(except reactor coolant pump (RCP) seal water injection or leakoff),

;

collection systems or

2. ~~b.~~ Leakage into the containment atmosphere from sources that are both specifically located and known ~~either not~~ to interfere with the operation of leakage detection systems ~~or not to be PRESSURE BOUNDARY LEAKAGE,~~ or

;

3. ~~c.~~ Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary~~-~~to~~-~~secondary leakage)~~.~~

not

;

b. Unidentified LEAKAGE see CTS 1.34 Markup

c. Pressure Boundary LEAKAGE see CTS 1.22 Markup

A01

A05

A01

1.1 DEFINITIONS

INSERVICE TESTING PROGRAM

INSERVICE TESTING PROGRAM

~~1.16~~ The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

A01

~~MEMBER(S) OF THE PUBLIC~~

~~1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.~~

A04

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 THE OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

A04

See ITS 5.5.1

OPERABLE – OPERABILITY

OPERABLE – OPERABILITY

~~1.19~~ A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s)~~,~~ and when all necessary attendant instructions, controls, electrical power, cooling ~~or~~ seal water, lubrication ~~or~~ other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

, and

, normal or emergency

safety

specified safety

and

A07

A08

A01

A08

MODE

~~OPERATIONAL MODE –~~ MODE

~~1.20~~ A~~n OPERATIONAL~~ MODE ~~(i.e., MODE)~~ shall correspond to any one inclusive combination of core reactivity condition, power level ~~and~~ average reactor coolant temperature specified in Table ~~1.2~~.

,

,

and reactor vessel head closure bolt tensioning

1.1-1 with fuel in the reactor vessel

A01

PHYSICS TESTS

PHYSICS TESTS

~~1.21~~ PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation ~~and (1)~~ described in Chapter 14.0 of the FSAR, ~~(2)~~ authorized under the provisions of 10 CFR 50.59, or ~~(3)~~ otherwise approved by the Commission.

. These tests are:

a.

, Initial Test Program

b.

c.

Nuclear Regulatory

A06

A01

1.1 **DEFINITIONS**

LEAKAGE

~~PRESSURE BOUNDARY~~ LEAKAGE

c.

an RCS

LEAKAGE

~~1.22~~ PRESSURE BOUNDARY LEAKAGE ~~shall be leakage~~ (except primary-to-secondary leakage) through a ~~non-isolable~~ fault in ~~a Reactor Coolant System~~ component body, pipe wall, or vessel wall.

,

LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.

A14

A01

A05

~~PROCESS CONTROL PROGRAM (PCP)~~

~~1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~

A04

~~PURGE – PURGING~~

~~1.24 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.~~

RATED THERMAL POWER

RATED THERMAL POWER

(RTP)

RTP

~~1.25 RATED THERMAL POWER~~ shall be a total reactor core heat transfer rate to the reactor coolant of 3020 MWt.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

PROTECTION

(RPS)

REACTOR ~~TRIP~~ SYSTEM RESPONSE TIME

RPS

s

RPS

that

~~1.26~~ The ~~REACTOR TRIP SYSTEM~~ RESPONSE TIME shall be ~~the~~ time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power to the CEA drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC~~.~~

, or the components have been evaluated in accordance with an NRC approved methodology.

L02

A01

~~REPORTABLE EVENT~~

~~1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.~~

A04

SHIELD BUILDING INTEGRITY

A04

1.28 SHIELD BUILDING INTEGRITY shall exist when:

See ITS 3.6.7

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

See ITS 3.6.9

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

1.1 DEFINITIONS

SHUTDOWN MARGIN

SHUTDOWN MARGIN

SDM

SDM

~~1.29~~ ~~SHUTDOWN MARGIN~~ shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all ~~full-length~~ ~~control element assemblies~~ (shutdown and regulating) are fully inserted except for the single ~~assembly~~ of highest reactivity worth which is assumed to be fully withdrawn.

A01

See CTS 4.1.1.1.1.a Markup

A09

CEA

CEAs

Insert 4

L03

~~SITE BOUNDARY~~

~~1.30 SITE BOUNDARY means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.~~

~~SOURCE CHECK~~

~~1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.~~

A04

~~STAGGERED TEST BASIS~~

~~1.32 A STAGGERED TEST BASIS shall consist of:~~

~~a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and~~

~~b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.~~

THERMAL POWER

THERMAL POWER

~~1.33~~ THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

A01

LEAKAGE

~~UNIDENTIFIED~~ LEAKAGE

A01

(except RCP seal water injection or leakoff) that

b.

~~1.34~~ UNIDENTIFIED LEAKAGE ~~shall be~~ all leakage ~~which~~ is not IDENTIFIED LEAKAGE ~~or CONTROLLED LEAKAGE.~~

A05

A04

; and

~~UNRESTRICTED AREA~~

~~1.35 Unrestricted area means an area, access to which is neither limited nor controlled by the licensee.~~

A04

L03

INSERT 4

However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation.

1.1 DEFINITIONS

~~UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F~~~~r~~

~~1.36 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.~~

A04

~~VENTILATION EXHAUST TREATMENT SYSTEM~~

~~1.37 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Features (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.~~

A04

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours
D	At least once per 24 hours
W	At least once per 7 days
4/M*	At least 4 per month at intervals of no greater than 9 days and a minimum of 48 per year
M	At least once per 31 days
Q	At least once per 92 days
SA	At least once per 184 days
R	At least once per 18 months
S/U	Prior to each reactor startup
P**	Completed prior to each release
SFCP	In accordance with the Surveillance Frequency Control Program
N.A.	Not applicable

LA02

* For Radioactive Effluent Sampling.

** For Radioactive Batch Releases only.

TABLE ~~1.2~~

1.1-1 (page 1 of 1)

~~OPERATIONAL~~ MODES

	~~OPERATIONAL~~ MODE	REACTIVITY CONDITION, K_{eff}	% OF RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE (°F)
1.	POWER OPERATION	≥ 0.99	> 5~~%~~	NA → ≥ ~~325°F~~
2.	STARTUP	≥ 0.99	≤ 5~~%~~	NA → ≥ ~~325°F~~
3.	HOT STANDBY	< 0.99	NA → ~~0~~	≥ 325~~°F~~
4.	HOT SHUTDOWN (b)	< 0.99	NA → ~~0~~	325~~°F~~ > Tavg > 200~~°F~~
5.	COLD SHUTDOWN (b)	< 0.99	NA → ~~0~~	≤ 200~~°F~~
6.	REFUELING** (c)	NA → ~~< 0.95~~	NA → ~~0~~	NA → ≤ 140~~°F~~

A01 A11 LA03

(a) * Excluding decay heat.

(c) ** ~~Fuel in the reactor vessel with the~~ One or more reactor vessel head closure bolts less than fully tensioned ~~or with the head removed~~.

(b) All reactor vessel head closure bolts fully tensioned.

A01 A06

Add proposed ITS Sections
1.2 Logical Connectors
1.3 Completion Times
1.4 Frequency

A10

ITS

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

See ITS 3.1.1

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN outside the COLR limits, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

SHUTDOWN MARGIN

capable of being

With any

s

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. ~~If the inoperable~~ CEA~~is~~ not fully inserted, ~~and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable,~~ the ~~above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn~~ worth of ~~the immovable or untrippable~~ CEA~~s~~.

A01

reactivity

these

must be accounted for in the determination of SDM

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

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* See Special Test Exception 3.10.1.

ITS

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

See ITS 3.1.1

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN outside the COLR limits, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

not capable of being fully inserted

With any

s

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. ~~If the inoperable~~ CEA ~~is immovable or untrippable~~, the ~~above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn~~ worth of ~~the immovable or untrippable~~ CEA~~(s)~~.

A01

SHUTDOWN MARGIN

these

must be accounted for in the determination of SDM

reactivity

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2b and by verifying at least two charging pumps are rendered inoperable by racking out their motor circuit breakers.

DISCUSSION OF CHANGES
ITS 1.0, USE AND APPLICATIONS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 1.4 defines a CHANNEL CALIBRATION as "the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated." ITS defines a CHANNEL CALIBRATION as "the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step." This results in a number of changes to the CTS.

- The CTS definition states, "The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions and shall include the CHANNEL FUNCTIONAL TEST." The ITS states, "The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST."

  This change is acceptable because the statements are equivalent in that both require that all needed portions of the channel be tested. The ITS definition reflects the CTS understanding that the CHANNEL CALIBRATION includes only those portions of the channel needed to perform the safety function including the required devices associated with the CHANNEL FUNCTIONAL TEST.

- The ITS adds the statement, "Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in place qualitative assessment of sensor behavior and normal

calibration of the remaining adjustable devices in the channel." The purpose of a CHANNEL CALIBRATION is to adjust the channel output so that the channel responds within the necessary range and accuracy to known values of the parameters that the channel monitors.

This change is acceptable because resistance temperature detectors and thermocouples are designed such that they have a fixed input/output response, which cannot be adjusted or changed once installed. Calibration of a channel containing an RTD or thermocouple is performed by applying the RTD or thermocouple fixed input/output relationship to the remainder of the channel, and making the necessary adjustments to the adjustable devices in the remainder of the channel to obtain the necessary output range and accuracy. Therefore, unlike other sensors, an RTD or thermocouple is not actually calibrated. The ITS CHANNEL CALIBRATION allowance for channels containing RTDs and thermocouples is consistent with the CTS calibration practices of these channels. This information is included in the ITS to avoid confusion but does not change the current CHANNEL CALIBRATION practices for these types of channels.

- The ITS adds the statement, "and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step."

  This change is acceptable because it results in no technical changes to the Technical Specifications. The CTS have implemented the Surveillance Frequency Control Program.

These changes are designated as administrative because they do not result in a technical change to the Technical Specifications.

A03 CTS Section 1.0 defines CHANNEL FUNCTIONAL TEST as "the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions." ITS Section 1.1 defines it as "the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step." This changes the CHANNEL FUNCTIONAL TEST by stating that the CHANNEL FUNCTIONAL TEST may include “the injection of a simulated or actual signal”, applies to “all devices in the channel required for OPERABILITY”, may “be performed by means of any series of sequential, overlapping, or total channel steps”, and “each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.” The addition of use of an actual signal is discussed in DOC L01.

- The CTS definition states that the CHANNEL FUNCTIONAL TEST shall verify that the channel is OPERABLE "including alarm and/or trip functions." Similarly, the ITS requirement states that the CHANNEL FUNCTIONAL TEST

shall verify OPERABILITY of "all devices in the channel required for channel OPERABILITY."

This change is acceptable because the statements are equivalent in that both require verification of channel OPERABILITY. The CTS and the ITS use different examples of what is included in a channel, but this does not change the intent of the requirement. The ITS use of the phrase "all devices in the channel required for channel OPERABILITY" reflects the CTS understanding that the test includes only those portions of the channel needed to perform the specified safety function(s).

- The ITS states "The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps."

  This change is acceptable because it states current Industry practice and is consistent with the current implementation of the CHANNEL FUNCTIONAL TEST. Therefore, this change does not result in a technical change to the Technical Specifications.

- The ITS adds the statement, "and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step."

  This change is acceptable because it results in no technical changes to the Technical Specifications. The CTS have implemented the Surveillance Frequency Control Program.

These changes are designated as administrative because they do not result in a technical change to the Technical Specifications.

A04 CTS Section 1.0 includes the following definitions:

- CONTAINMENT VESSEL INTEGRITY
- CONTROLLED LEAKAGE
- CORE ALTERATION
- FREQUENCY NOTATION
- GASEOUS RADWASTE TREATMENT SYSTEM
- MEMBER(S) OF THE PUBLIC
- OFFSITE DOSE CALCULATION MANUAL (ODCM)
- PROCESS CONTROL PROGRAM (PCP)
- PURGE – PURGING
- REPORTABLE EVENT
- SHIELD BULIDING INTEGRITY
- SITE BOUNDARY
- SOURCE CHECK
- STAGGERED TEST BASIS
- UNRESTRICTED AREA
- UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r
- VENTILATION EXHAUST TREATMENT SYSTEM (PSL Unit 2 Only)

The ITS does not use this terminology and ITS Section 1.1 does not contain these definitions. The FREQUENCY NOTATION definition deletion is discussed in DOC LA02. The VENTILATION EXHAUST TREATMENT SYSTEM definition deletion is only applicable to PSL Unit 2 because this definition is not contained in the PSL Unit 1 TS.

These changes are acceptable because the terms are not used as defined terms in the ITS. Discussions of any technical changes related to the deletion of these terms are included in the DOCs for the CTS sections in which the terms are used. These changes are designated as administrative because they eliminate defined terms that are no longer used.

A05 CTS Section 1.0 provides definitions for IDENTIFIED LEAKAGE, PRESSURE BOUNDARY LEAKAGE, and UNIDENTIFIED LEAKAGE. ITS Section 1.1 includes these requirements in one definition called LEAKAGE (which includes three categories: identified LEAKAGE, unidentified LEAKAGE, and pressure boundary LEAKAGE). This changes the CTS by incorporating the definitions into the ITS LEAKAGE definition with no technical changes.

This change is acceptable because it results in no technical changes to the Technical Specifications. This change is designated an administrative change in that it rearranges existing definitions, with no change in intent.

A06 CTS Section 1.20 and Table 1.2, OPERATIONAL MODES" provide a description of the MODES. CTS Table 1.2 contains Note ** that states, "Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed." ITS Section 1.1 and Table 1.1-1, "MODES," changes the CTS MODE definitions in the following ways:

- The CTS Table 1.2 Note ** condition "fuel in the vessel" is moved to the ITS MODE definition.

  This change is acceptable because it moves information within the Technical Specifications with no change in intent. Each MODE in the Table includes fuel in the vessel.

- CTS Table 1.1, Note ** in part states, "…with the vessel head closure bolts less than fully tensioned or with the head removed." ITS splits this portion of the Note into two Notes, Notes (b), and (c). ITS Note (b) states, "All reactor vessel head closure bolts fully tensioned," while Note (c) states, "One or more reactor vessel head closure bolts less than fully tensioned." This change simplifies what CTS is stating by clearly defining when the reactor is in a refueling condition instead of a shutdown condition.

  This change is acceptable because the revised phrase is consistent with the current interpretation and usage. MODE 6 is currently declared when the first vessel head closure bolt is detensioned. This change also eliminates a redundant phrase. The reactor vessel head cannot be removed unless the reactor vessel head closure bolts are unbolted and they cannot be unbolted

unless they are detensioned. Since “vessel head closure bolts less then fully tensioned” is already specified in the CTS Note, including “or with the reactor head removed” is unnecessary.

- ITS Table 1.1-1 contains a new Note b, which applies to MODES 4 and 5. Note b states "All reactor vessel head closure bolts fully tensioned." This Note is the opposite of CTS Note ** and ITS Table 1.1-1 Note (c).

  This change is acceptable because it avoids a conflict between the definition of MODE 6 and the other MODES should RCS temperature increase above the CTS MODE 6 temperature limit while a reactor vessel head closure bolt is less than fully tensioned. This ITS Note is included only for clarity. It is consistent with the current use of MODES 4 and 5 and does not result in any technical change to the application of the MODES.

- For consistency with the Notes in ITS Table 1.1-1, the ITS definition of MODE adds, "reactor vessel head closure bolt tensioning" to the list of characteristics that define a MODE. Currently, the CTS definition does not include this clarification.

  This change is acceptable because the definition of MODE should be consistent with the MODE table in order to avoid confusion. This change is made only for consistency and results in no technical changes to the Technical Specifications.

These changes are designated as administrative because they clarify the application of the MODES and no technical changes to the MODE definitions are made. The clarifications are consistent with the current use and application of the MODES.

A07 The CTS Section 1.0 definition of OPERABLE - OPERABILITY requires that “all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).” The ITS Section 1.1 definition of OPERABLE - OPERABILITY will replace the phrase “electrical power" with "normal or emergency electrical power." This changes the CTS definition of OPERABLE - OPERABILITY by allowing a device to be considered OPERABLE with either normal or emergency power available.

The OPERABILITY requirements for normal and emergency power sources are addressed in CTS Section 3/4.8. These requirements allow only the normal or the emergency electrical power source to be OPERABLE, provided its redundant system(s), subsystem(s), train(s), component(s), and device(s) (redundant to the systems, subsystems, trains, components, and devices with an inoperable power source) are OPERABLE. Therefore, the ITS definition uses the words "normal or emergency" consistent with CTS Section 3/4.8. This change is designated administrative since the ITS definition is effectively the same as the CTS definition.

A08 The CTS Section 1.0 definition of OPERABLE - OPERABILITY requires that “A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).” The ITS Section 1.1 definition of OPERABLE - OPERABILITY will replace the phrase “specified function(s)” with "specified safety functions" and will replace the phrase “function(s)” with "specified safety functions."

The CTS Section 1.0 definition of OPERABLE - OPERABILITY requires a system, subsystem, train, component, or device to be capable of performing its "specified function(s)" and all necessary support systems to also be capable of performing their "function(s)." The ITS Section 1.1 definition of OPERABLE - OPERABILITY requires the system, subsystem, train, component, or device to be capable of performing the "specified safety function(s)," and requires all necessary support systems that are required for the system, subsystem, train, component, or device to perform its "specified safety function(s)" to also be capable of performing their related support functions. This changes the CTS by altering the requirement to be able to perform "functions" to a requirement to be able to perform "safety functions."

The purpose of the CTS and ITS definitions of OPERABLE - OPERABILITY are to ensure that the safety analysis assumptions regarding equipment and variables are valid. This change is acceptable because the intent of both the CTS and ITS definitions is to address the safety function(s) assumed in the accident analysis and not encompass other non-safety functions a system may also perform. These non-safety functions are not assumed in the safety analysis and are not needed in order to protect the public health and safety. This change is consistent with the current interpretation and use of the terms OPERABLE and OPERABILITY. This change is designated as administrative as it does not change the current use and application of the Technical Specifications.

A09 CTS Section 1.0 provides a definition of SHUTDOWN MARGIN (SDM). The ITS Section 1.1 definition of SDM contains the following difference from the CTS definition.

- The CTS definition of SDM does not include a statement requiring an increased allowance for the withdrawn worth of an immovable or untrippable CEA(s). This requirement is contained in CTS 4.1.1.1.1.a and CTS 4.1.1.2.a. The ITS definition of SDM includes this increased allowance by stating, “ With any CEAs not capable of being fully inserted, the reactivity worth of these CEAs must be accounted for in the determination of SDM.” This changes the CTS definition of SDM to include the requirement in CTS 4.1.1.1.1.a and CTS 4.1.1.2.a for an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

This change is acceptable because it is consistent with the existing SDM requirements associated with a CEA not capable of being fully inserted as specified in CTS 3.1.1.1 and 3.1.1.2.

This change is designated as administrative because it does not represent a technical change to the Technical Specifications

A10 ITS Sections 1.2, 1.3, and 1.4 contain information that is not in the CTS. This change to the CTS adds explanatory information on ITS usage that is not applicable to the CTS. The added sections are:

- Section 1.2 - Logical Connectors

  Section 1.2 provides specific examples of the logical connectors "AND" and "OR" and the numbering sequence associated with their use.

- Section 1.3 - Completion Times

  Section 1.3 provides guidance on the proper use and interpretation of Completion Times. The section also provides specific examples that aid in the use and understanding of Completion Times

- Section 1.4 – Frequency

  Section 1.4 provides guidance on the proper use and interpretation of Surveillance Frequencies. The section also provides specific examples that aid in the use and understanding of Surveillance Frequency.

This change is acceptable because it aids in the understanding and use of the format and presentation style of the ITS. The addition of these sections does not add or delete technical requirements, and will be discussed specifically in those Technical Specifications where application of the added sections results in a change. This change is designated as administrative because it does not result in a technical change to the Technical Specifications.

A11 CTS Table 1.2, OPERATIONAL MODES, is revised. The corresponding table in ITS Section 1.1 is Table 1.1-1, MODES. The changes to the CTS are:

- CTS Table 1.2 minimum average reactor coolant temperature for MODES 1 and 2 is changed from ≥ 325°F to "NA" (not applicable) in ITS Table 1.1-1.

  This change is acceptable because ITS LCO 3.4.2, RCS Minimum Temperature for Criticality, provides the minimum reactor coolant temperature limits for MODES 1 and 2. Therefore, the 325°F minimum temperature does not provide any useful information in ITS Table 1.1-1, and is deleted from the CTS.

- CTS Table 1.2 MODE 6 upper limit on average reactor coolant temperature (< 140°F) is removed. In ITS Table 1.1-1, the MODE 6 average reactor coolant temperature limit is specified as "NA" (not applicable).

This change is acceptable because it eliminates a conflict in the CTS MODE Table. If the average coolant temperature exceeds the upper limit with the reactor vessel head closure bolts less than fully tensioned, the CTS Table could be misinterpreted as no MODE being applicable. This is not the intent of the CTS or ITS MODE 6 definitions. By removing the temperature reference, this ambiguity is eliminated.

- CTS Table 1.2 % RATED THERMAL POWER limit of 0% for MODES 3, 4, 5, and 6 is changed in ITS Table 1.1-1 to "NA" (not applicable).

  This change is acceptable because the reactivity and plant equipment limitations in MODES 3, 4, 5, and 6 do not allow power operation. Therefore, it is not necessary to have these restrictions in the MODE Table.

- CTS Table 1.2 contains the unit designators of percent (%) and degrees Fahrenheit (°F) next to the values. This is changed in ITS Table 1.1-1 by removing the designator from the individual value(s).

  This change is acceptable because the designators are contained in the labels associated with the columns. Therefore, it is not necessary to have these designators in the MODE Table.

These changes are designated as administrative because they result in no technical changes to the Technical Specifications.

A12 CTS 1.0 states, "The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications." The Note for ITS Section 1.1 states, "The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases." The ITS Note adds a clarification phrase that the defined terms also apply to the Bases.

The ITS Section 1.0 Note serves the same purpose as the CTS 1.0 statement. ITS Section 1.1 Note clarifies that the defined terms also apply to the Bases. This change is consistent with formatting requirements in the ISTS and is consistent with the current use. This change is designated as administrative because it does not represent a technical change to the Technical Specifications.

A13 **Unit 1 only:** CTS Section 1.0 provides a definition of SHUTDOWN MARGIN (SDM). An *note to Unit 1 CTS 4.1.1.1.1.e.2 modifies the CTS definition by stating that for Modes 3 and 4, during calculation of shutdown margin with all CEA's verified fully inserted, the single CEA with the highest reactivity worth need not be assumed to be stuck in the fully withdrawn position. The definition of SDM in ITS Section 1.1 incorporates the aspect of the CTS *note by clarifying, “However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation.”

This changes the CTS by embedding the *note in Unit 1 CTS 4.1.1.1.1.e.2 into the SDM definition. Although not explicitly stated, the intent of the Unit 1 CTS

*note is based on the CEA position indicating system design for plants with a Combustion Engineering nuclear steam supply system, which includes independent and diverse methods of verifying CEA position. Therefore, the ITS definition of SDM clarifies this intent of verifying the CEAs are fully inserted by two independent means.

This change is designated as administrative because it does not represent a technical change to the method of calculating SDM when the CEAs are verified fully inserted by two independent means.

Unit 2 CTS related to SDM does not contain a clarifying note like the *note in Unit 1 CTS 4.1.1.1.1.e.2 and the *note in the Unit 1 CTS is limited to Modes 3 and 4. The equivalent change to the Unit 2 SDM definition and the expanded allowance for Unit 1 SDM definition to include all modes with CEAs inserted are discussed in DOC L03.

A14 CTS 1.22, PRESSURE BOUNDARY LEAKAGE, states that PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary-to-secondary LEAKAGE) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall." The ITS definition states that "Pressure Boundary LEAKAGE is LEAKAGE (except primary to secondary LEAKAGE) through a fault in an RCS component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE." This changes the CTS by revising the pressure boundary LEAKAGE definition to delete the word "non-isolable and adding the sentence "LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE." These changes are consistent with Technical Specification Task Force (TSTF) traveler TSTF-554-A, Revision 1, "Revise Reactor Coolant Leakage Requirements." As stated in the NRC Safety Evaluation (SE) accompanying TSTF-554-A, deletion of the word "nonisolable" does not alter the fundamental meaning that pressure boundary leakage represents degradation that could ultimately result in a loss of structural integrity. The NRC concluded that deleting the word "nonisolable" provides a clearer definition of pressure boundary leakage and does not conflict with the reactor coolant pressure boundary definition in 10 CFR 50.2. Additionally, the sentence "LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE," is consistent with the discussion in NRC Regulatory Guide 1.45 that pressure boundary leakage is not leakage from seals, gaskets, or packing. That considered, the changes to the definition of pressure boundary leakage are acceptable.

TSTF traveler TSTF-554-A incorporated changes to the Standard Technical Specifications (STSs) under the consolidated line item improvement process (CLIIP). TSTF-554-A was approved for use by the NRC as documented in the accompanying SE dated December 18, 2020 (ADAMS Accession No. ML20322A361, ML20322A024). PSL has reviewed the NRC SE and concluded that the justification presented in TSTF-554-A and the SE prepared by the NRC staff are applicable to PSL and justify this change. TSTF-554-A revised the technical specifications related to the LEAKAGE definition. Related changes are made to ITS 3.4.13, RCS Operational LEAKAGE, as discussed in ITS 3.4.13 DOC L01.

This change is designated as administrative because it provides clarification and does not represent a technical change to the Technical Specifications.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* The ITS definition for AXIAL SHAPE INDEX moves the discussion and equation regarding modification of the trip signals in the reactor protection system to determine the core axial power distribution for a given channel to the COLR. The AXIAL SHAPE INDEX curves are provided in the COLR.

CTS 2.2.1 requires reactor protective instrumentation setpoints be set consistent with the Trip Setpoint values shown in Table 2.2-1, Reactor Protective Instrumentation Trip Setpoint Limits. The Applicability is for each channel in Table 3.3-1, Reactor Protective Instrumentation. CTS 3.2.1 Action references COLR Figure 3.2-2, “AXIAL SHAPE INDEX vs. Maximum Allowable Power Level” and CTS LCO 3.2.5.d references COLR Figure 3.2-4 “AXIAL SHAPE INDEX Operating Limits vs. THERMAL POWER.”

ITS LCO 3.2.1 requires that the LHR shall not exceed the limits specified in the COLR, and ITS LCO 3.2.5 requires that the ASI shall not exceed the limits specified in the COLR.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains AXIAL SHAPE INDEX regarding modification of the trip signals in the reactor protection system to determine the core axial power distribution for a given channel. Also, this change is acceptable because these types of procedural details will be adequately controlled in the COLR. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program).* CTS 1.13 and CTS Table 1.1 state FREQUENCY NOTATION for the performance of Surveillance Requirements in the CTS. The ITS state periodic Frequency as "In accordance with the Surveillance Frequency Control Program." The CTS have implemented the Surveillance Frequency Control Program. This changes the CTS by deleting the FREQUENCY NOTATION definition and moving the FREQUENCY NOTATION Table to the Surveillance Frequency Control Program.

The removal of FREQUENCY NOTATION for the performance of Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement by stating "In accordance with the Surveillance Frequency Control Program". This change is acceptable because these types of procedural details will be adequately controlled in the Surveillance Frequency Control Program. This change is designated as a less restrictive removal of detail change because the FREQUENCY NOTATION for the performance of Surveillance Frequencies is being removed from the Technical Specifications and placed in a license control document.

LA03 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 1.2, OPERATIONAL MODES, states that MODE 6 is restricted to reactivity conditions with $k_{eff} \leq 0.95$. ITS Table 1.1-1, MODES, does not contain this restriction.

This change is acceptable because the core reactivity requirements for MODE 6 are covered in ITS 3.9.1, "Boron Concentration," by requiring the boron concentration in the Reactor Coolant System to be maintained within the limits specified in the COLR. The ITS LCO section of the 3.9.1 Bases states "The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations." Moving this detail from the MODE Table to the LCO 3.9.1 Bases eliminates the potential to misinterpret the MODE table and not apply the MODE 6 requirements if the reactor vessel head closure bolts are less than fully tensioned, fuel is in the reactor vessel, and core reactivity exceeds a keff of 0.95. ITS LCO 3.9.1 will ensure that the appropriate reactivity conditions are maintained in MODE 6, so it is not necessary to have this restriction in the MODE Table in order to provide adequate protection of the public health and safety. Once moved to the Bases, any changes to the core reactivity requirement will be controlled by the Technical Specifications Bases Control Program described in Chapter 5 of the ITS. This change is designated a less restrictive removal of detail because it moves information from the Technical Specifications to the Bases.

LESS RESTRICTIVE CHANGES

L01 CTS Section 1.0 definition of CHANNEL FUNCTIONAL TEST requires the use of a simulated signal when performing the test. ITS Section 1.1 allows the use of a

simulated or actual signal when performing the test. This changes the CTS by allowing the use of unplanned actuations to perform the Surveillance based on the collection of sufficient information to satisfy the surveillance test requirements.

This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal. Therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change is designated as less restrictive because it allows an actual signal to be credited for Surveillance where only a simulated signal was previously allowed.

L02 CTS Section 1.0 definition of Engineered Safety Feature (ESF) Response Time and Reactor Protection System (RPS) response time require that in lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC. ITS Section 1.1 allows components to be evaluated in accordance with an NRC approved methodology.

The change revises the CTS definition for ESF Response Time and RPS response time that are referenced in Surveillance Requirements (SRs), hereafter referred to as response time testing (RTT). The definitions are revised to add the statement “or the components have been evaluated in accordance with an NRC approved methodology” at the end of the last sentence in the definitions. The changes are based on Technical Specifications Task Force (TSTF) traveler TSTF-569, Revision 2, “Revise Response Time Testing Definition,” dated June 25, 2019 (ADAMS Accession No. ML19176A034). The NRC issued a final safety evaluation (SE) approving TSTF-569, Revision 2, on August 14, 2019 (ADAMS Accession No. ML19176A191). | R1

This change is acceptable because the changes are based on TSTF-569, Revision 2 and there are no proposed variations from the TS changes described in TSTF-569, Revision 2. This change is designated as less restrictive because it allows use of additional NRC approved methodology for evaluation of RTT.

L03 **Unit 2 only:** CTS Section 1.0 definition of SHUTDOWN MARGIN (SDM) requires that the calculation assume that all CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth which is assumed to be fully withdrawn. The ITS definition allows that with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation.

This change revises the Technical Specifications to include the allowance that with all CEAs verified inserted by two independent means, it is not necessary to account for a stuck CEA in determination of SDM. Consistent with the Combustion Engineering nuclear steam supply system design, the PSL Unit 1 and Unit 2 design include two independent and diverse CEA position indicating systems. The independent systems are the pulse counting CEA position indication system and the reed switch CEA position indication system.

This change is acceptable because the CTS definition was developed considering a worst case condition to assure sufficient SDM is available. However, if all rods can be verified to be fully inserted, the worst case condition is not applicable and the requirement to assume a rod is stuck in the fully withdrawn position is overly conservative and unnecessary to assure safe plant operation. The provision of the ITS exception that requires the rod positions to be verified by two independent means provides adequate assurance that all rods are fully inserted and that SDM may be calculated without the conservative assumption of a fully withdrawn rod. PSL Unit 1 and Unit 2 design provide two independent systems to verify all CEAs are fully inserted. This change is designated as less restrictive because it allows for an exception to the CTS requirements.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

1.0 USE AND APPLICATION

1.1 Definitions

----------NOTE----------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

	Term	Definition
1.1	ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
1.2	AXIAL SHAPE INDEX (ASI)	ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core. ASI = (LOWER - UPPER) / (LOWER + UPPER)
	~~AZIMUTHAL POWER TILT (T_q) - Digital~~	~~AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.~~
1.3	AZIMUTHAL POWER TILT (T_q) ~~- Analog~~	AZIMUTHAL POWER TILT shall be the maximum of the difference between the power generated in any core quadrant (upper or lower) (P_{quad}) and the average power of all quadrants (P_{avg}) in that half (upper or lower) of the core, divided by the average power of all quadrants in that half (upper or lower) of the core. T_q = Max $\lvert (P_{quad} - P_{avg}) / P_{avg} \rvert$
1.4	CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps~~[~~, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step~~]~~.

1.1 Definitions

1.5	CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
1.6	CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be~~:~~ ~~a. Analog and bistable channels –~~ the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY~~, and~~ ~~b. Digital computer channels - the use of diagnostic programs to test digital computer hardware and the injection of simulated process data into the channel to verify OPERABILITY of all devices in the channel required for channel OPERABILITY.~~ The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps~~[~~, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step~~]~~.
1.9a	CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.
1.10	DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using ~~---------- Reviewer's Note ----------~~ ~~The first set of thyroid dose conversion factors shall be used for plants licensed to 10 CFR 100.11. The following Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) conversion factors shall be used for plants licensed to 10 CFR 50.67.~~ ~~----------~~

1.1 Definitions

1.10 DOSE EQUIVALENT I-131 (continued)

~~[thyroid dose conversion factors from:~~

~~a. Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or~~

~~b. Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or~~ 3

~~c. ICRP-30, 1979, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," or~~

~~d. Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."~~

~~OR~~

Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11.~~]~~ 3

1.11 DOSE EQUIVALENT XE-133

DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides ~~[~~Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138~~]~~ actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using ~~[~~effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil" ~~or the average gamma disintegration energies as provided in ICRP Publication 38, "Radionuclide Transformations" or similar source]~~. 2 4

1.12 ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where

1.1 Definitions

1.12 ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME (continued)

applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.

1.16 INSERVICE TESTING PROGRAM

The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

1.15 LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known to not interfere with the operation of leakage detection systems; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

1.34

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; and

c. Pressure Boundary LEAKAGE

1.22

LEAKAGE (except primary to secondary LEAKAGE) through a fault in an RCS component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.

1.1 Definitions

1.20	MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
1.19	OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
1.21	PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter [14, Initial Test Program] of the FSAR, (2) b. Authorized under the provisions of 10 CFR 50.59, or c. Otherwise approved by the Nuclear Regulatory Commission.
	~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~	~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4.~~ (5)
1.25	RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~[3410]~~ 3020 MWt. (2)

1.1 Definitions

1.26	REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.
1.29	SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming~~:~~ ~~a.~~ All ~~full length~~ CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. With any CEAs not capable of being fully inserted, the reactivity worth of these CEAs must be accounted for in the determination of SDM~~, and~~ ~~[b. There is no change in part length CEA position.]~~
	~~[STAGGERED TEST BASIS~~	~~A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.]~~
1.33	THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	325 ≥ ~~[350]~~ (2)
4	Hot Shutdown[(b)]	< 0.99	NA	~~[350]~~ > T_{avg} > ~~[~~200~~]~~ (2)
5	Cold Shutdown[(b)]	< 0.99	NA	≤ ~~[~~200~~]~~ (2)
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.
EXAMPLES	The following examples illustrate the use of logical connectors.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Unless otherwise specified, the Completion Time begins when a senior licensed operator on the operating shift crew with responsibility for plant operations makes the determination that an LCO is not met and an ACTIONS Condition is entered. The "otherwise specified" exceptions are varied, such as a Required Action Note or Surveillance Requirement Note that provides an alternative time to perform specific tasks, such as testing, without starting the Completion Time. While utilizing the Note, should a Condition be applicable for any reason not addressed by the Note, the Completion Time begins. Should the time allowance in the Note be exceeded, the Completion Time begins at that point. The exceptions may also be incorporated into the Completion Time. For example, LCO 3.8.1, "AC Sources - Operating," Required Action B.2, requires declaring required feature(s) supported by an inoperable diesel generator, inoperable when the redundant required feature(s) are inoperable. The Completion Time states, "4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)." In this case the Completion Time does not begin until the conditions in the Completion Time are satisfied. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability. If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate

1.3 Completion Times

DESCRIPTION (continued)

Completion Times are tracked for each Condition starting from the discovery of the situation that required entry into the Condition, unless otherwise specified.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition, unless otherwise specified.

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ."

1.3 Completion Times

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

1.3 Completion Times

EXAMPLES (continued)

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

1.3 Completion Times

EXAMPLES (continued)

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

It is possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

1.3 Completion Times

EXAMPLES (continued)

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

EXAMPLE 1.3-5

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

1.3 Completion Times

EXAMPLES (continued)

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

1.3 Completion Times

EXAMPLES (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated. AND A.2 Restore subsystem to OPERABLE status.	1 hour AND Once per 8 hours thereafter 72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

1.3 Completion Times

EXAMPLES (continued)

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

---------- Reviewer's Note ----------
Example 1.3-8 is only applicable to plants that have adopted the Risk Informed Completion Time Program.
---------- 6

[EXAMPLE 1.3-8

2

ACTIONS

6

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

1.3 Completion Times

EXAMPLES (continued)

When a subsystem is declared inoperable, Condition A is entered. The 7 day Completion Time may be applied as discussed in Example 1.3-2. However, the licensee may elect to apply the Risk Informed Completion Time Program which permits calculation of a Risk Informed Completion Time (RICT) that may be used to complete the Required Action beyond the 7 day Completion Time. The RICT cannot exceed 30 days. After the 7 day Completion Time has expired, the subsystem must be restored to OPERABLE status within the RICT or Condition B must also be entered.

The Risk Informed Completion Time Program requires recalculation of the RICT to reflect changing plant conditions. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

If the 7 day Completion Time clock of Condition A has expired and subsequent changes in plant condition result in exiting the applicability of the Risk Informed Completion Time Program without restoring the inoperable subsystem to OPERABLE status, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start.

If the RICT expires or is recalculated to be less than the elapsed time since the Condition was entered and the inoperable subsystem has not been restored to OPERABLE status, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable subsystems are restored to OPERABLE status after Condition B is entered, Condition A is exited, and therefore, the Required Actions of Condition B may be terminated. ~~]~~ 1

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0~~.2~~, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements. (7) Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillances, or both. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction. The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria. Some Surveillances contain Notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

1.4 Frequency

DESCRIPTION (continued)

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be performed until 12 hours after ≥ 25% RTP. ----------	
Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Only required to be met in MODE 1. ----------	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Only required to be performed in MODE 1. ----------	
Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be met in MODE 3. ----------	
Verify parameter is within limits.	24 hours

Example 1.4-~~[~~6~~]~~ specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

1

1.0 USE AND APPLICATION

1.1 Definitions

----------NOTE----------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

	Term	Definition
1.1	ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
1.2	AXIAL SHAPE INDEX (ASI)	ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core. ASI = (LOWER - UPPER) / (LOWER + UPPER)
	~~AZIMUTHAL POWER TILT (T_q) - Digital~~	~~AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.~~
1.3	AZIMUTHAL POWER TILT (T_q) ~~- Analog~~	AZIMUTHAL POWER TILT shall be the maximum of the difference between the power generated in any core quadrant (upper or lower) (P_{quad}) and the average power of all quadrants (P_{avg}) in that half (upper or lower) of the core, divided by the average power of all quadrants in that half (upper or lower) of the core. T_q = Max $\lvert (P_{quad} - P_{avg}) / P_{avg} \rvert$
1.4	CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps~~[~~, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step~~]~~.

1.1 Definitions

1.5 **CHANNEL CHECK**

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

1.6 **CHANNEL FUNCTIONAL TEST**

A CHANNEL FUNCTIONAL TEST shall be~~:~~

~~a. Analog and bistable channels – the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY; and~~

~~b. Digital computer channels - the use of diagnostic programs to test digital computer hardware and the injection of simulated process data into the channel to verify OPERABILITY of all devices in the channel required for channel OPERABILITY.~~

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps~~[~~, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step~~]~~.

1.9a **CORE OPERATING LIMITS REPORT (COLR)**

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.

1.10 **DOSE EQUIVALENT I-131**

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using

~~---------- Reviewer's Note ----------~~
~~The first set of thyroid dose conversion factors shall be used for plants licensed to 10 CFR 100.11. The following Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) conversion factors shall be used for plants licensed to 10 CFR 50.67.~~
~~----------~~

1.1 Definitions

1.10 DOSE EQUIVALENT I-131 (continued)

~~[thyroid dose conversion factors from:~~

~~a. Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or~~

~~b. Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or~~ 3

~~c. ICRP-30, 1979, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," or~~

~~d. Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."~~

~~OR~~

Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11.~~]~~ 3

1.11 DOSE EQUIVALENT XE-133

DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides ~~[~~Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138~~]~~ actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using ~~[~~effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil" ~~or the average gamma disintegration energies as provided in ICRP Publication 38, "Radionuclide Transformations" or similar source]~~. 2 4

1.12 ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where

1.1 Definitions

1.12 ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME (continued)

applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.

1.16 INSERVICE TESTING PROGRAM

The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

1.15 LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known to not interfere with the operation of leakage detection systems; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

1.34 b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; and

1.22 c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a fault in an RCS component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.

1.1 Definitions

1.20	MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
1.19	OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
1.21	PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter ~~[~~14, Initial Test Program~~]~~ of the FSAR, (2) b. Authorized under the provisions of 10 CFR 50.59, or c. Otherwise approved by the Nuclear Regulatory Commission.
	~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~	~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4.~~ (5)
1.25	RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~[3410]~~ 3020 MWt. (2)

1.1 Definitions

1.26	REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.
1.29	SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming~~:~~ ~~a.~~ All ~~full length~~ CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. With any CEAs not capable of being fully inserted, the reactivity worth of these CEAs must be accounted for in the determination of SDM~~, and~~ ~~[b. There is no change in part length CEA position.]~~
	~~[STAGGERED TEST BASIS~~	~~A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.]~~
1.33	THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	325 ≥ ~~[350]~~ (2)
4	Hot Shutdown[(b)]	< 0.99	NA	~~[350]~~ > T_{avg} > ~~[~~200~~]~~ (2)
5	Cold Shutdown[(b)]	< 0.99	NA	≤ ~~[~~200~~]~~ (2)
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.
EXAMPLES	The following examples illustrate the use of logical connectors.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Unless otherwise specified, the Completion Time begins when a senior licensed operator on the operating shift crew with responsibility for plant operations makes the determination that an LCO is not met and an ACTIONS Condition is entered. The "otherwise specified" exceptions are varied, such as a Required Action Note or Surveillance Requirement Note that provides an alternative time to perform specific tasks, such as testing, without starting the Completion Time. While utilizing the Note, should a Condition be applicable for any reason not addressed by the Note, the Completion Time begins. Should the time allowance in the Note be exceeded, the Completion Time begins at that point. The exceptions may also be incorporated into the Completion Time. For example, LCO 3.8.1, "AC Sources - Operating," Required Action B.2, requires declaring required feature(s) supported by an inoperable diesel generator, inoperable when the redundant required feature(s) are inoperable. The Completion Time states, "4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)." In this case the Completion Time does not begin until the conditions in the Completion Time are satisfied. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability. If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate

1.3 Completion Times

DESCRIPTION (continued)

Completion Times are tracked for each Condition starting from the discovery of the situation that required entry into the Condition, unless otherwise specified.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition, unless otherwise specified.

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ."

1.3 Completion Times

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

1.3 Completion Times

EXAMPLES (continued)

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

1.3 Completion Times

EXAMPLES (continued)

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

It is possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

1.3 Completion Times

EXAMPLES (continued)

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

EXAMPLE 1.3-5

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

1.3 Completion Times

EXAMPLES (continued)

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

1.3 Completion Times

EXAMPLES (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated. AND A.2 Restore subsystem to OPERABLE status.	1 hour AND Once per 8 hours thereafter 72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

1.3 Completion Times

EXAMPLES (continued)

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

---------- Reviewer's Note ----------
Example 1.3-8 is only applicable to plants that have adopted the Risk Informed Completion Time Program.
---------- 6

[EXAMPLE 1.3-8

2

ACTIONS

6

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

1.3 Completion Times

EXAMPLES (continued)

When a subsystem is declared inoperable, Condition A is entered. The 7 day Completion Time may be applied as discussed in Example 1.3-2. However, the licensee may elect to apply the Risk Informed Completion Time Program which permits calculation of a Risk Informed Completion Time (RICT) that may be used to complete the Required Action beyond the 7 day Completion Time. The RICT cannot exceed 30 days. After the 7 day Completion Time has expired, the subsystem must be restored to OPERABLE status within the RICT or Condition B must also be entered.

The Risk Informed Completion Time Program requires recalculation of the RICT to reflect changing plant conditions. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

If the 7 day Completion Time clock of Condition A has expired and subsequent changes in plant condition result in exiting the applicability of the Risk Informed Completion Time Program without restoring the inoperable subsystem to OPERABLE status, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start.

If the RICT expires or is recalculated to be less than the elapsed time since the Condition was entered and the inoperable subsystem has not been restored to OPERABLE status, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable subsystems are restored to OPERABLE status after Condition B is entered, Condition A is exited, and therefore, the Required Actions of Condition B may be terminated. ~~]~~

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0~~.2~~, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements. (7) Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillances, or both. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction. The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria. Some Surveillances contain Notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

1.4 Frequency

DESCRIPTION (continued)

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be performed until 12 hours after ≥ 25% RTP. ----------	
Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Only required to be met in MODE 1. ----------	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Only required to be performed in MODE 1. ----------	
Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be met in MODE 3. ----------	
Verify parameter is within limits.	24 hours

Example 1.4-~~[~~6~~]~~ specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

1

JUSTIFICATION FOR DEVIATIONS
ITS 1.0, USE AND APPLICATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS definition for DOSE EQUIVALENT I-131 includes a Reviewer's Note. The Reviewer's Note has been deleted. The Note informs the NRC Reviewer what is needed to meet this requirement. The Reviewer's Note is not meant to be retained in the final version of the ITS. St. Lucie Plant (PSL) Unit 1 and Unit 2 are licensed to 10 CFR 50.67.

4. The ISTS definition for DOSE EQUIVALENT XE-133 contains bracketed information with two options for “effective dose conversion factors for air submission.” PSL Unit 1 and Unit 2 are licensed to the effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil" Therefore, this option will be maintained.

5. PSL does not propose to use a PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) and will not relocate the Pressure and Temperature limits from the Technical Specifications. The current limits will be retained in the ITS. Therefore, the definition of PTLR has not been incorporated into the ITS.

6. PSL ITS 1.3, Example 1.3-8 Reviewer’s Note states “Example 1.3-8 is only applicable to plants that have adopted the Risk Informed Completion Time Program.” PSL Unit 1 has implemented the Risk Informed Completion Time Program as approved in License Amendment No. 247 (ADAMS Accession No. ML19113A099). PSL Unit 2 has implemented the Risk Informed Completion Time Program as approved in License Amendment No. 199 (ADAMS Accession No. ML19113A099). Therefore, the Reviewer’s Note is deleted, and the Example 1.3-8 brackets are deleted.

7. Typographical error is corrected. The proper section for Surveillance Requirement (SR) Applicability is Section 3.0.

8. The ISTS definition of SHUTDOWN MARGIN deletes “full length” as related to control element assemblies (CEAs). Part length CEAs are not used at PSL Unit 1 and PSL Unit 2.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L01

Florida Power & Light Company (FPL) is converting St. Lucie Plant Unit 1 and Unit 2 to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5, "Standard Technical Specifications, Combustion Engineering Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of no significant hazards considerations for conversion to NUREG-1432.

The CTS Section 1.0 definition of CHANNEL FUNCTIONAL TEST requires the use of a simulated signal when performing the test. ITS Section 1.1 allows the use of a simulated or actual signal when performing the test. This changes the CTS by allowing the use of unplanned actuations to perform the Surveillance based on the collection of sufficient information to satisfy the surveillance test requirements.

FPL has evaluated whether or not a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the CHANNEL FUNCTIONAL TEST. This change allows taking credit for unplanned actuations if sufficient information is collected to satisfy the surveillance test requirements. This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal, and the proposed requirement does not change the technical content or validity of the test. This change will not affect the probability of an accident. The source of the signal sent to components during a Surveillance is not assumed to be an initiator of any analyzed event. The consequence of an accident is not affected by this change. The results of the testing, and, therefore, the likelihood of discovering an inoperable component, are unaffected. As a result, the assurance that equipment will be available to mitigate the consequences of an accident is unaffected.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the CHANNEL FUNCTIONAL TEST. This change will not

physically alter the plant (no new or different type of equipment will be installed). The change does not require any new or revised operator actions.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the CHANNEL FUNCTIONAL TEST. The margin of safety is not affected by this change. This change allows taking credit for unplanned actuations if sufficient information is collected to satisfy the surveillance test requirements. This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal. As a result, the proposed requirement does not change the technical content or validity of the test.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L02

Florida Power & Light Company (FPL) is converting St. Lucie Plant Unit 1 and Unit 2 to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5, "Standard Technical Specifications, Combustion Engineering Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of no significant hazards considerations for conversion to NUREG-1432.

The change revises the CTS definitions for Engineered Safety Feature (ESF) Response Time and Reactor Protection System (RPS) response time that are referenced in Surveillance Requirements (SRs), hereafter referred to as response time testing (RTT). The definitions are revised to add the statement “or the components have been evaluated in accordance with an NRC approved methodology” at the end of the last sentence in the definitions. The changes are based on Technical Specifications Task Force (TSTF) traveler TSTF-569, Revision 2, “Revise Response Time Testing Definition,” dated June 25, 2019 (ADAMS Accession No. ML19176A034). The NRC issued a final safety evaluation (SE) approving TSTF-569, Revision 2, on August 14, 2019 (ADAMS Accession No. ML19176A191). FPL is not proposing any variations from the TS changes described in TSTF-569, Revision 2, or the applicable parts of the NRC SE of TSTF-569, Revision 2.

FPL has evaluated whether or not a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change revises the CTS Definition of RPS and ESF instrumentation response time to permit FPL to evaluate using an NRC-approved methodology and apply a bounding response time for some components in lieu of measurement. The requirement for the instrumentation to actuate within the response time assumed in the accident analysis is unaffected.

   The response time associated with the RPS and ESF instrumentation is not an initiator of any accident. Therefore, the proposed change has no significant effect on the probability of any accident previously evaluated.

   The affected RPS and ESF instrumentation are assumed to actuate their respective components within the required response time to mitigate accidents previously evaluated. Revising the TS definition for RPS and ESF instrumentation response times to allow an NRC-approved methodology for verifying response time for some components does not alter the surveillance requirements that verify the RPS and ESF instrumentation response times are within the required limits. As such, the TS will continue to assure that the RPS and ESF instrumentation actuate their

associated components within the specified response time to accomplish the required safety functions assumed in the accident analyses.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change revises the CTS Definition of RPS and ESF instrumentation response time to permit FPL to evaluate using an NRC-approved methodology and apply a bounding response time for some components in lieu of measurement. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change does not alter any assumptions made in the safety analyses. The proposed change does not alter the limiting conditions for operation for the RPS or ESF instrumentation, nor does it change the Surveillance Requirement to verify the RPS and ESF instrumentation response times are within the required limits. As such, the proposed change does not alter the operability requirements for the RPS and ESF instrumentation, and therefore, does not introduce any new failure modes.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change revises the CTS Definition of RPS and ESF instrumentation response time to permit the licensee to evaluate using an NRC-approved methodology and apply a bounding response time for some components in lieu of measurement. The proposed change has no effect on the required RPS and ESF instrumentation response times or setpoints assumed in the safety analyses and the TS requirements to verify those response times and setpoints. The proposed change does not alter any Safety Limits or analytical limits in the safety analysis. The proposed change does not alter the TS operability requirements for the RP] and ESF instrumentation. The RPS and ESF instrumentation actuation of the required systems and components at the required setpoints and within the specified response times will continue to accomplish the design basis safety functions of the associated systems and components in the same manner as before. As such, the RPS and ESF instrumentation will continue to perform the required safety functions as assumed in the safety analyses for all previously evaluated accidents.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L03

Florida Power & Light Company (FPL) is converting St. Lucie Plant Unit 1 and Unit 2 to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5, "Standard Technical Specifications, Combustion Engineering Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of no significant hazards considerations for conversion to NUREG-1432.

CTS Section 1.0 definition of SHUTDOWN MARGIN (SDM) requires that the calculation assume that all CEAs (shutdown and regulating) are fully inserted except for the single rod assembly of highest reactivity worth which is assumed to be fully withdrawn. The ITS definition allows that with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. This change revises the Technical Specifications to include the allowance that with all CEAs verified inserted by two independent means, it is not necessary to account for a stuck CEA in determination of SDM. PSL Unit 1 and Unit 2 design include two independent CEA position indicating systems. The independent systems are the pulse counting CEA position indication system and the reed switch CEA position indication system.

This change is acceptable because the CTS definition was developed considering a worst case condition to assure sufficient SDM is available. However, if all rods can be verified to be fully inserted, the worst case condition is not applicable and the requirement to assume a rod is stuck in the fully withdrawn position is overly conservative and unnecessary to assure safe plant operation. The provision of the ITS exception that requires the rod positions to be verified by two independent means provides adequate assurance that all rods are fully inserted and that SDM may be calculated without the conservative assumption of a fully withdrawn rod. PSL Unit 1 and Unit 2 design provide two independent systems to verify all CEAs are fully inserted. This change is designated as less restrictive because it allows for an exception to the CTS requirements.

FPL has evaluated whether or not a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   This change revises the Technical Specifications to include the allowance that with all CEAs verified inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. The proposed change adds an exception to the SDM definition. This change allows the SDM to be determined without assuming a rod is fully withdrawn when all CEAs are verified by two independent means to be fully inserted. The proposed change continues to assure adequate SDM is maintained consistent with the actual plant conditions. PSL Unit 1 and Unit 2 design

provide two independent systems to verify all CEAs inserted by two independent means. The proposed change will not affect the probability of an accident. The proposed change does not affect any accident initiators. The consequence of an accident is not affected by the proposed change because adequate SDM is maintained. The assumptions of accidents previously evaluated remain unaffected. Therefore, this change will not involve a significant increase in the probability or consequence of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change provides an exception to the SDM definition requirement to assume one CEA fully withdrawn. This change will not physically alter the plant (no new or different type of equipment will be installed). PSL Unit 1 and Unit 2 design provide two independent systems to verify all CEAs inserted by two independent means.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change provides an exception to the SDM definition requirement to assume one rod fully withdrawn when all rods are adequately verified to be fully inserted. PSL Unit 1 and Unit 2 design provide two independent systems to verify all CEAs inserted by two independent means. The proposed change continues to assure sufficient SDM is maintained consistent with the assumptions of the accident analyses. The proposed change does not affect the assumptions of any accident analysis and does not adversely affect the safe operation of the plant. As such, the margin of safety is not affected by this change.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

ENCLOSURE 2

VOLUME 4

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 2.0
SAFETY LIMITS

Revision 1

LIST OF ATTACHMENTS

1. ITS Chapter 2.0, Safety Limits

ATTACHMENT 1

ITS Chapter 2.0, Safety Limits

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

~~SECTION 2.0~~

~~SAFETY LIMITS~~

~~AND~~

~~LIMITING SAFETY SYSTEM SETTINGS~~

A01

A01

(SLs)

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~

SLs

2.1 **2.1 ~~SAFETY LIMITS~~**

2.1.1 **REACTOR CORE** SLs

2.1.1.1 2.1.1 The combination of THERMAL POWER, pressurizer pressure, and ~~maximum~~ cold leg coolant temperature shall not exceed the limits shown ~~on~~ in Figure 2.1-1.

the highest operating loop

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2.1.1.1 **~~APPLICABILITY~~:** In MODES 1 and 2~~.~~ ,

2.1.1.2 **~~ACTION~~:**

2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained within fuel melt temperature limits specified in the COLR.

M01

~~When the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line,~~ 2.2.1 be in ~~HOT STANDBY~~ within 1 hour.

MODE 3

SL

2.1.2 **REACTOR COOLANT SYSTEM PRESSURE**

be maintained ≤

2.1.2 The ~~Reactor Coolant System~~ pressure shall ~~not exceed~~ 2750 psia.

RCS

2.1.2 **~~APPLICABILITY~~:** In MODES 1, 2, 3, 4 and 5~~.~~ ,

2.2 **~~ACTION~~:**

2.2 SAFETY LIMIT VIOLATIONS

2.2.1 If SL 2.1.1 is violated, restore compliance and

2.2.2 If SL 2.1.2 is violated:

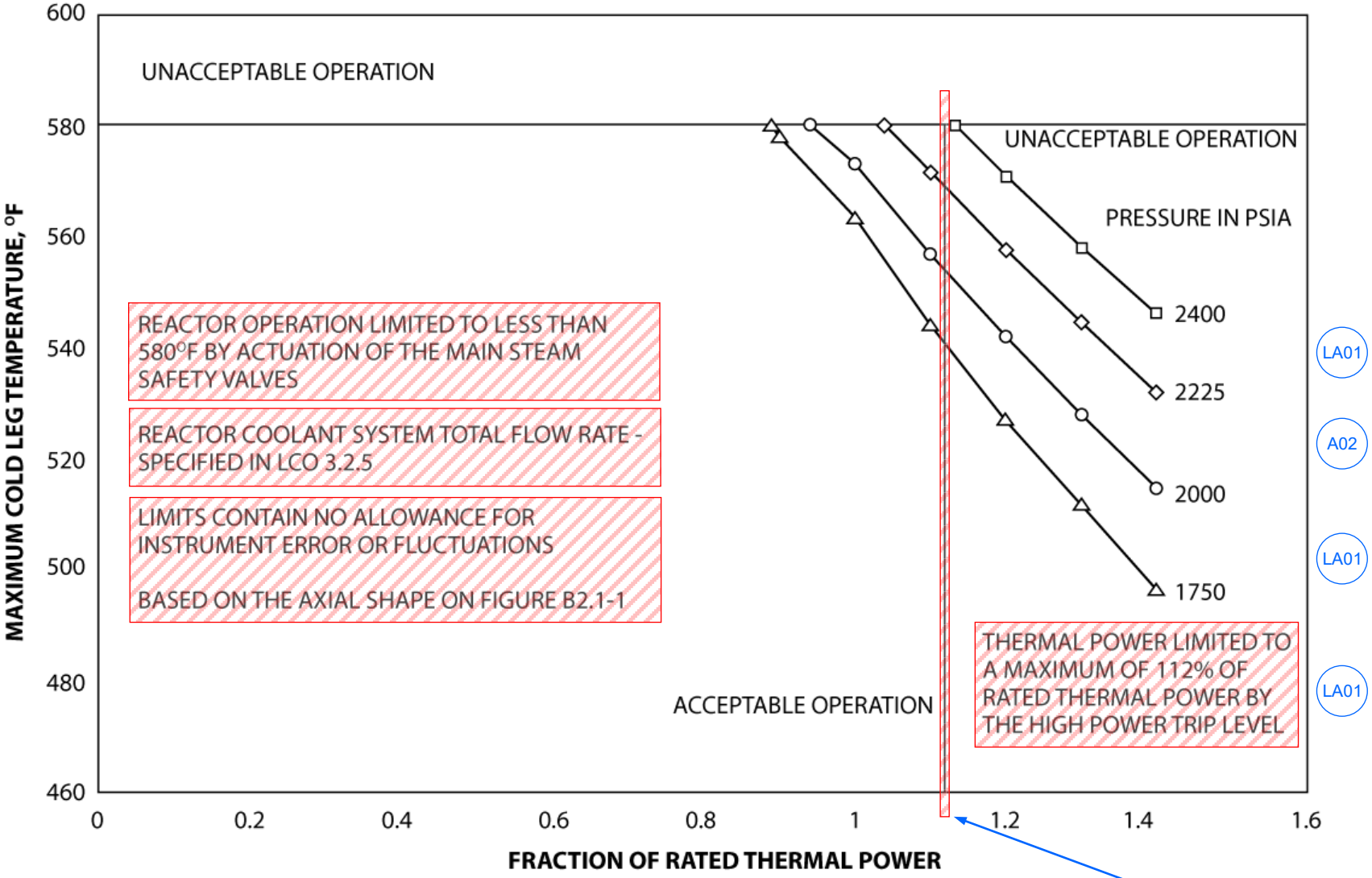
2.2.2.1 **MODE~~S~~ 1 ~~and~~ 2** In or , restore compliance and

~~Whenever the Reactor Coolant System pressure has exceeded 2750 psia,~~ 2.2.2.1 be in ~~HOT STANDBY~~ ~~with the Reactor Coolant System pressure within its limit~~ within 1 hour.

MODE 3

2.2.2.2 **MODE~~S~~ 3, 4 ~~and~~ 5** In or , restore compliance

~~Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit~~ 2.2.2.2 within 5 minutes.



LA01

A02

LA01

LA01

A03

Figure 2.1.1-1

FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT – ~~FOUR REACTOR COOLANT PUMPS OPERATING~~

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(page 1 of 1)

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

See ITS 3.3.1

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: AS SHOWN FOR EACH CHANNEL IN TABLE 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level – High (1) Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of < 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow – Low (1) Four Reactor Coolant Pumps Operating	≥ 95% of minimum reactor coolant flow with 4 pumps operating *	≥ 95% of minimum reactor coolant flow with 4 pumps operating *
4. Pressurizer Pressure – High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure – High	≤ 3.3 psig	≤ 3.3 psig
6. Steam Generator Pressure – Low (2)	≥ 600 psia	≥ 600 psia
7. Steam Generator Water Level – Low	≥ 35.0% Water Level – each steam generator	≥ 35.0% Water Level – each steam generator
8. Local Power Density – High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

See ITS 3.3.1

* For minimum reactor coolant flow with 4 pumps operating, refer to Technical Specification LCO 3.2.5.

See ITS 3.3.1

See ITS 3.3.1

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4
9a. Steam Generator Pressure Difference High (1) (logic in TM/LP)	≤ 135 psid	≤ 135 psid
10. Loss of Turbine – Hydraulic Fluid Pressure – Low (3)	≥ 800 psig	≥ 800 psig
11. Rate of Change of Power – High (4)	≤ 2.49 decades per minute	≤ 2.49 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is ≥ 1% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is ≥ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4} % and above 15% of RATED THERMAL POWER.

See ITS 3.3.1

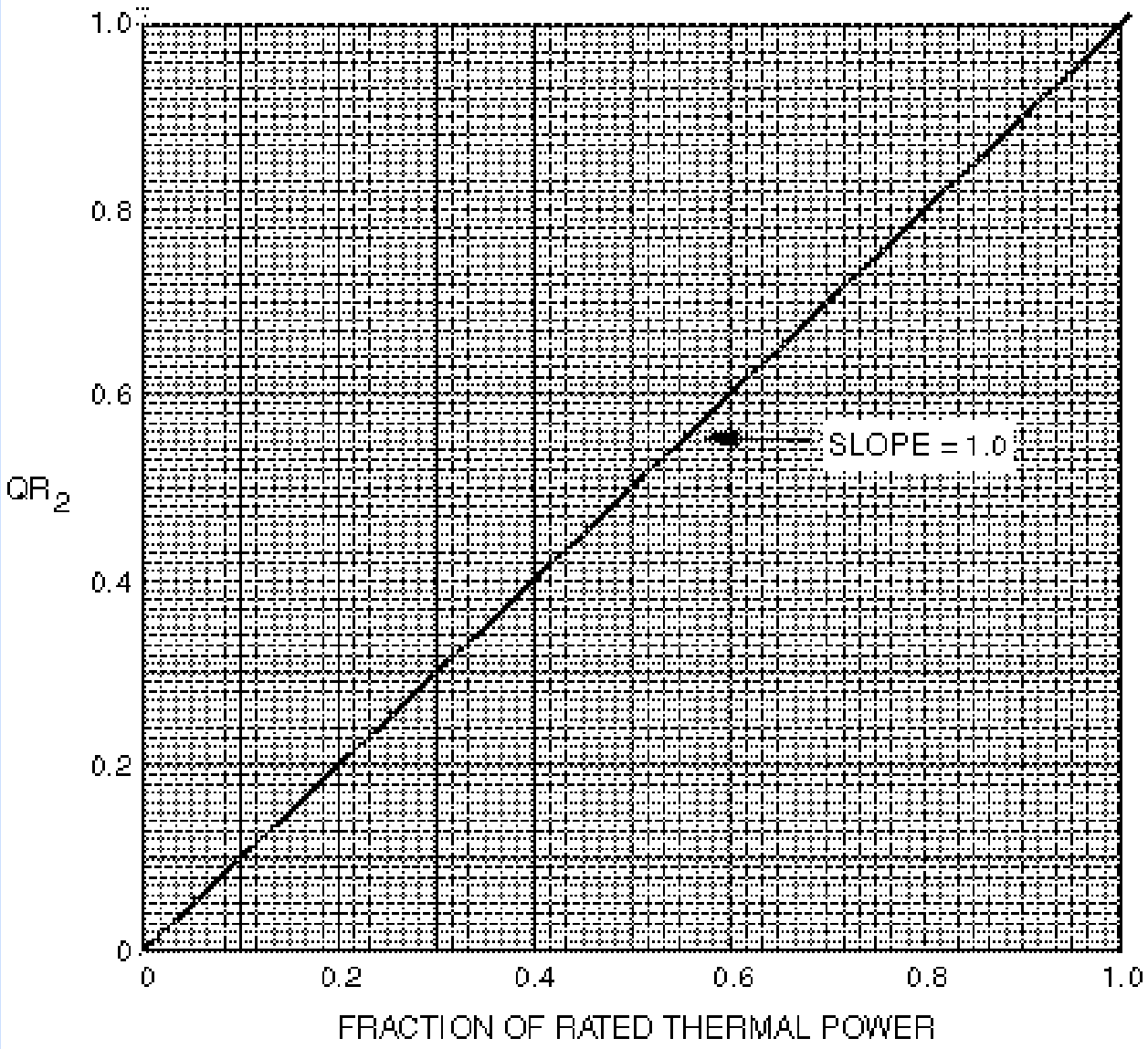
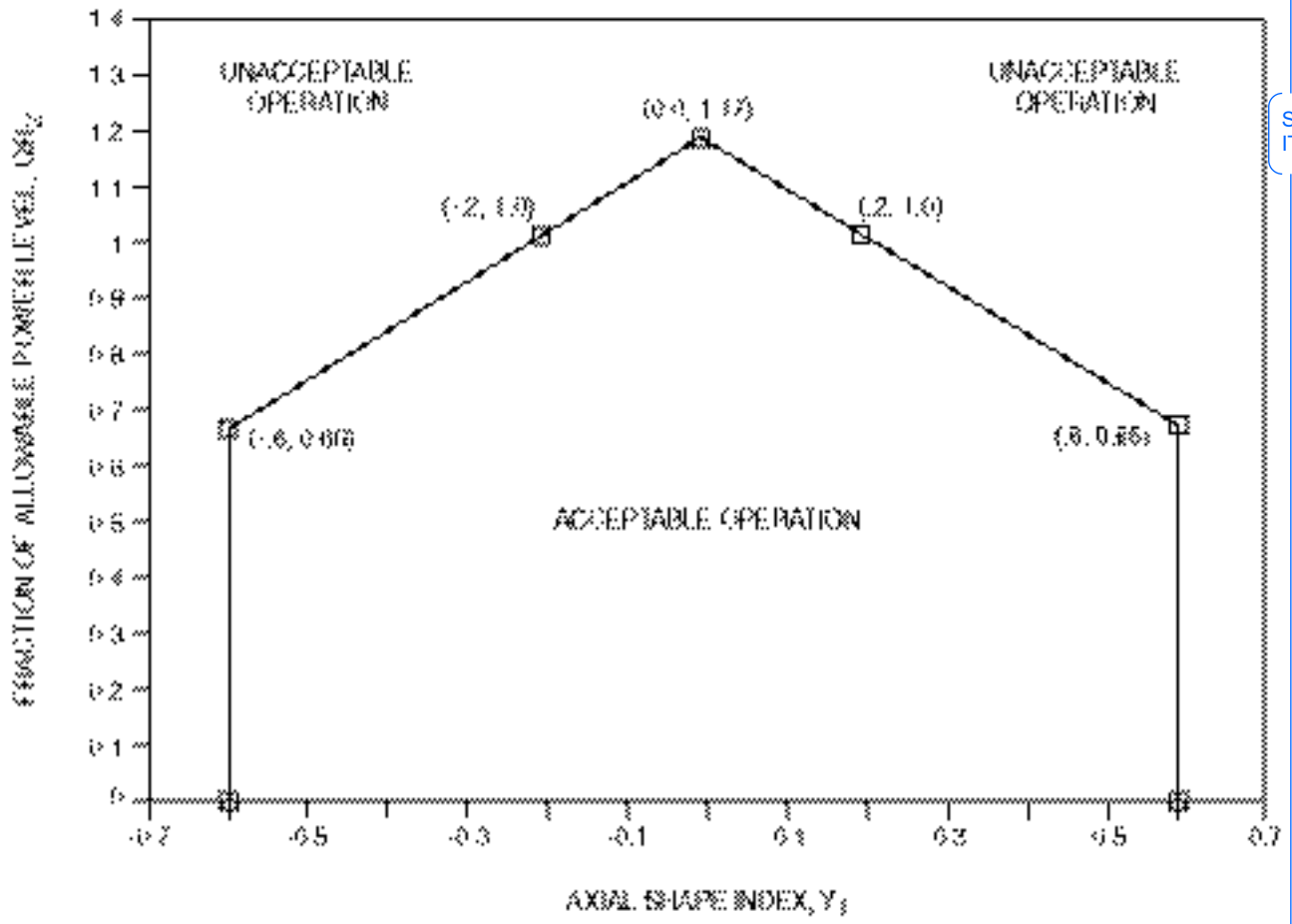


FIGURE 2.2-1

Local Power Density – High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)



See ITS 3.3.1

FIGURE 2.2-2

Local Power Density – High Trip Setpoint Part 2 (QR_2 Versus Y_1)

See ITS 3.3.1

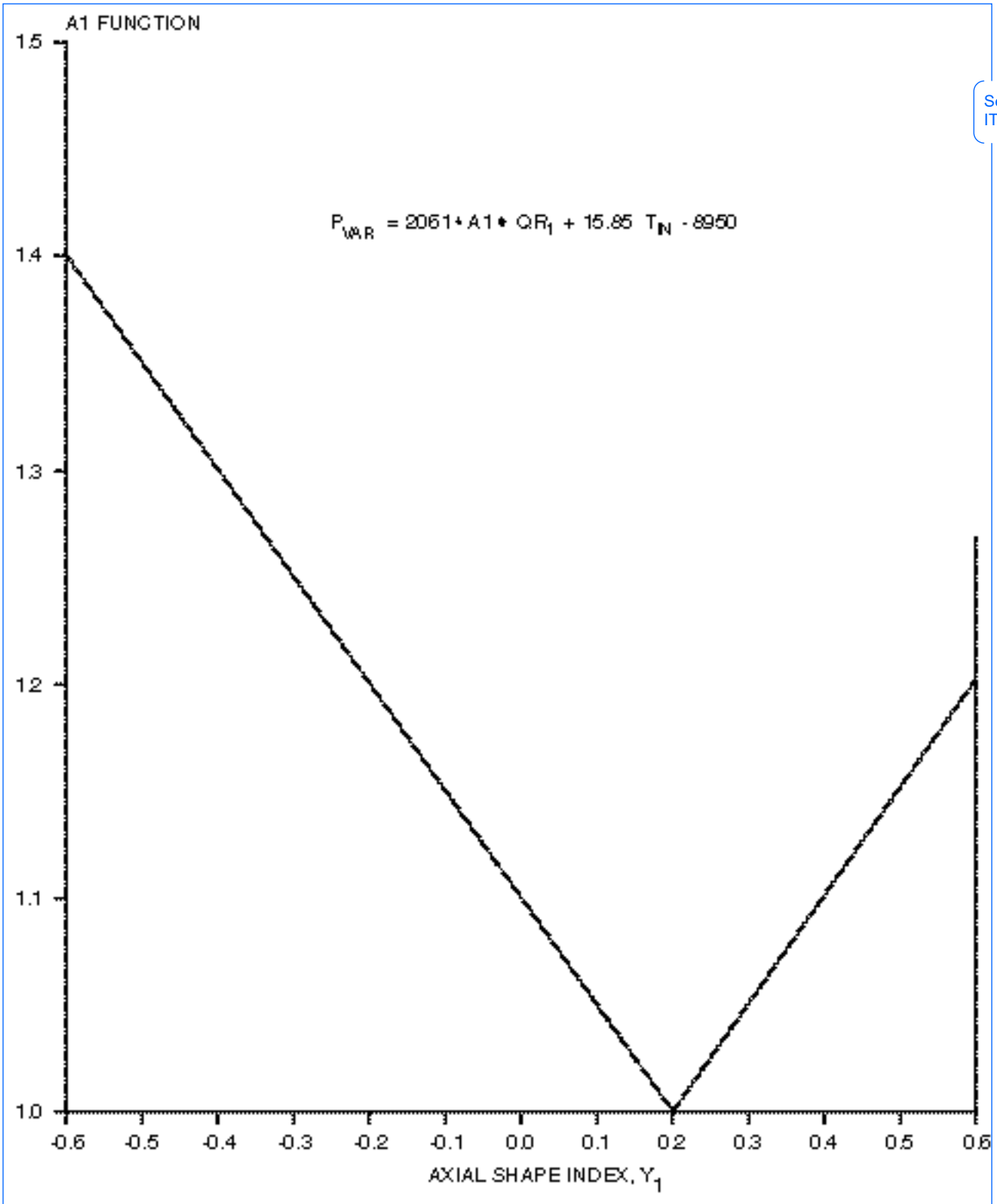
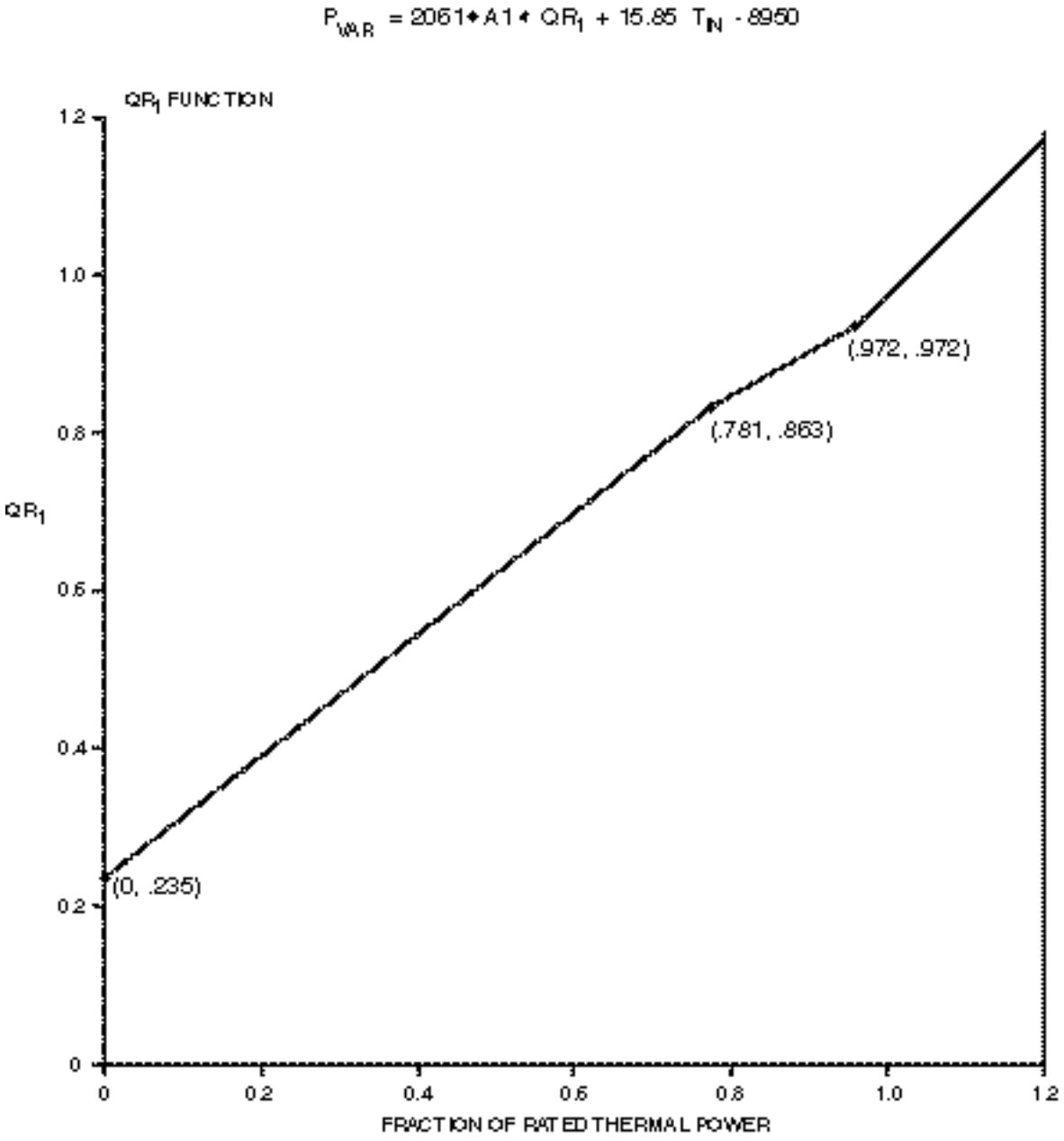


FIGURE 2.2-3

Thermal Margin/Low Pressure Trip Setpoint



See ITS 3.3.1

FIGURE 2.2-4

Thermal Margin/Low Pressure Trip Setpoint
Part 2 (Fraction of RATED THERMAL POWER Versus QR_1)

A01

(SLs)

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~

SLs

2.1 ## 2.1 ~~SAFETY LIMITS~~

SLs

2.1.1 ### 2.1.1 REACTOR CORE

~~DNBR~~

the highest operating loop

2.1.1.1 2.1.1.1 The combination of THERMAL POWER, pressurizer pressure, and ~~maximum~~ cold leg coolant temperature shall not exceed the limits shown ~~on~~ Figure 2.1-1.

in

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In

,

2.1.1.1 **~~APPLICABILITY~~**: MODES 1 and 2~~.~~

2.1.1.2 **~~ACTION:~~**

2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained below fuel melt temperature limits specified in the COLR.

M01

2.2.1 ~~Whenever the combination of THERMAL POWER, pressurizer pressure and maximum cold leg coolant temperature has exceeded the limits shown on Figure 2.1-1,~~ be in ~~HOT STANDBY~~ within 1 hour.

MODE 3

SL

2.1.2 ### REACTOR COOLANT SYSTEM PRESSURE

be maintained ≤

2.1.2 The ~~Reactor Coolant System~~ pressure shall ~~not exceed~~ 2750 psia.

RCS

In

,

2.1.2 **~~APPLICABILITY~~**: MODES 1, 2, 3, 4 and 5~~.~~

2.2 2.2 SAFETY LIMIT VIOLATIONS

~~ACTION:~~

2.2.1 If SL 2.1.1 is violated, restore compliance and

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 **MODE~~S~~ 1 ~~and~~ 2**

In

or

, restore compliance and

MODE 3

2.2.2.1 ~~Whenever the Reactor Coolant System pressure has exceeded 2750 psia,~~ be in ~~HOT STANDBY with the Reactor Coolant System pressure within its limit~~ within 1 hour.

2.2.2.2 **MODE~~S~~ 3, 4 ~~and~~ 5**

In

or

, restore compliance

2.2.2.2 ~~Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit~~ within 5 minutes.

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

See ITS 3.3.1

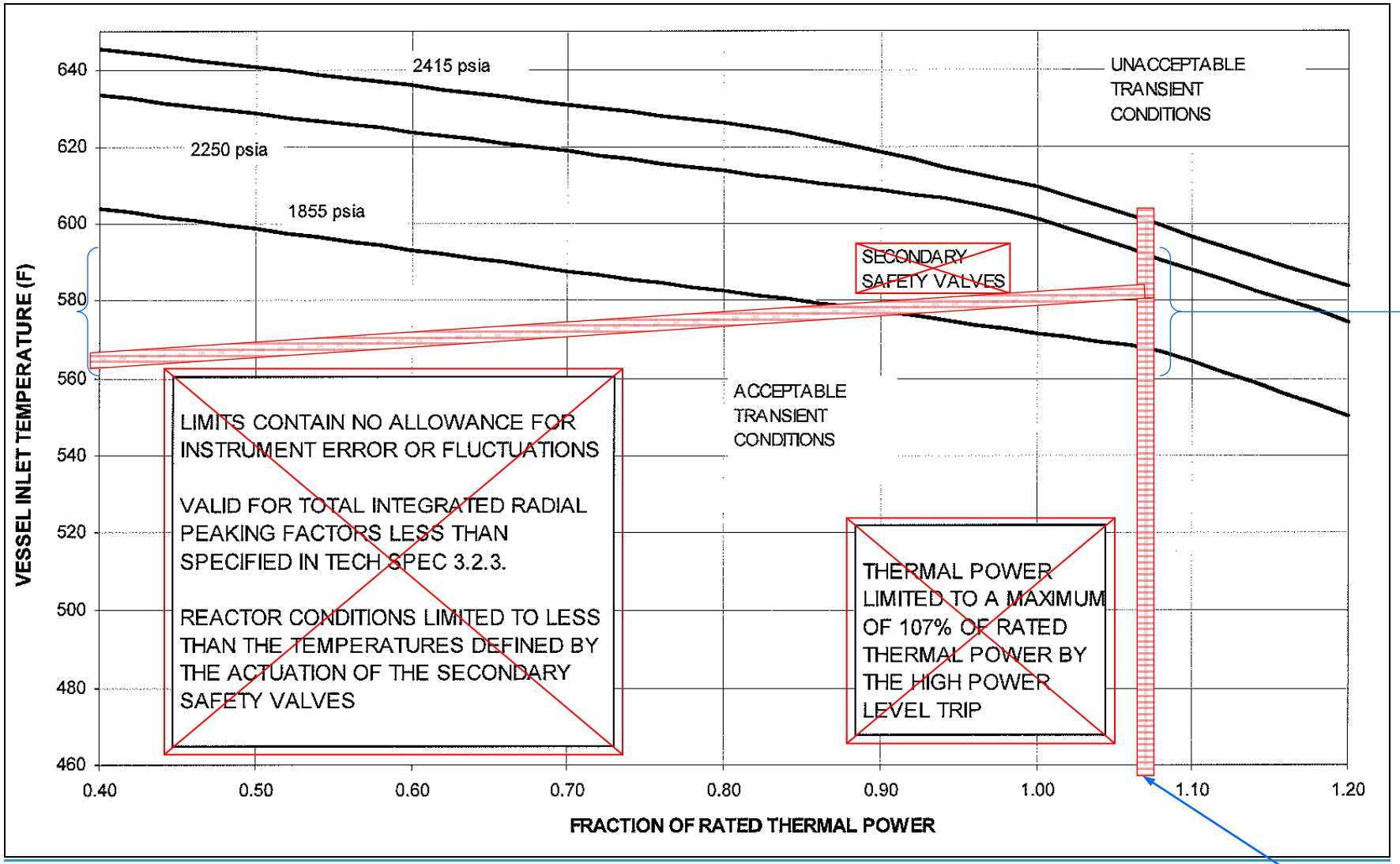
REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.



A03

LA01

A03

Figure 2.1.1-1

FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES
~~FOUR REACTOR COOLANT PUMPS OPERATING~~

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(page 1 of 1)

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Variable Power Level – High(1)		
Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Pressurizer Pressure – High	≤ 2370 psia	≤ 2374 psia
4. Thermal Margin/Low Pressure(1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.
5. Containment Pressure – High	≤ 3.0 psig	≤ 3.1 psig
6. Steam Generator Pressure – Low	≥ 626.0 psia(2) ~~(2)~~	≥ 621.0 psia(2) ~~(2)~~
7. Steam Generator Pressure Difference – High (Logic in TM/LP Trip Unit)(1)	≤ 120.0 psid	≤ 132.0 psid
8. Steam Generator Level – Low~~Low~~	≥ ~~20.5~~35.0%(3) ~~(3)~~	≥ ~~19.5~~35.0%(3) ~~(3)~~

See ITS 3.3.1

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density – High(5) Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Loss of Component Cooling Water to Reactor Coolant Pumps – Low	≥ 636 gpm**	≥ 636 gpm
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
13. Rate of Change of Power – High(4)	≤ 2.49 decades per minute	≤ 2.49 decades per minute
14. Reactor Coolant Flow – Low(1)	≥ 95.4% of minimum Reactor Coolant flow with four pumps operating*	≥ 94.9% of minimum Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure – Low(5)	≥ 800 psig	≥ 800 psig

See ITS 3.3.1

* For minimum reactor coolant flow with four pumps operating, refer to Technical Specification LCO 3.2.5.

** 10-minute time delay after relay actuation.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

See ITS 3.3.1

TABLE NOTATION

(1) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER during testing pursuant to Special Test Exception 3.10.3; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

(3) % of the narrow range steam generator level indication.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq$ 10^{-4}% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

(5) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is
greater than or equal to 15% of RATED THERMAL POWER.

See ITS 3.3.1

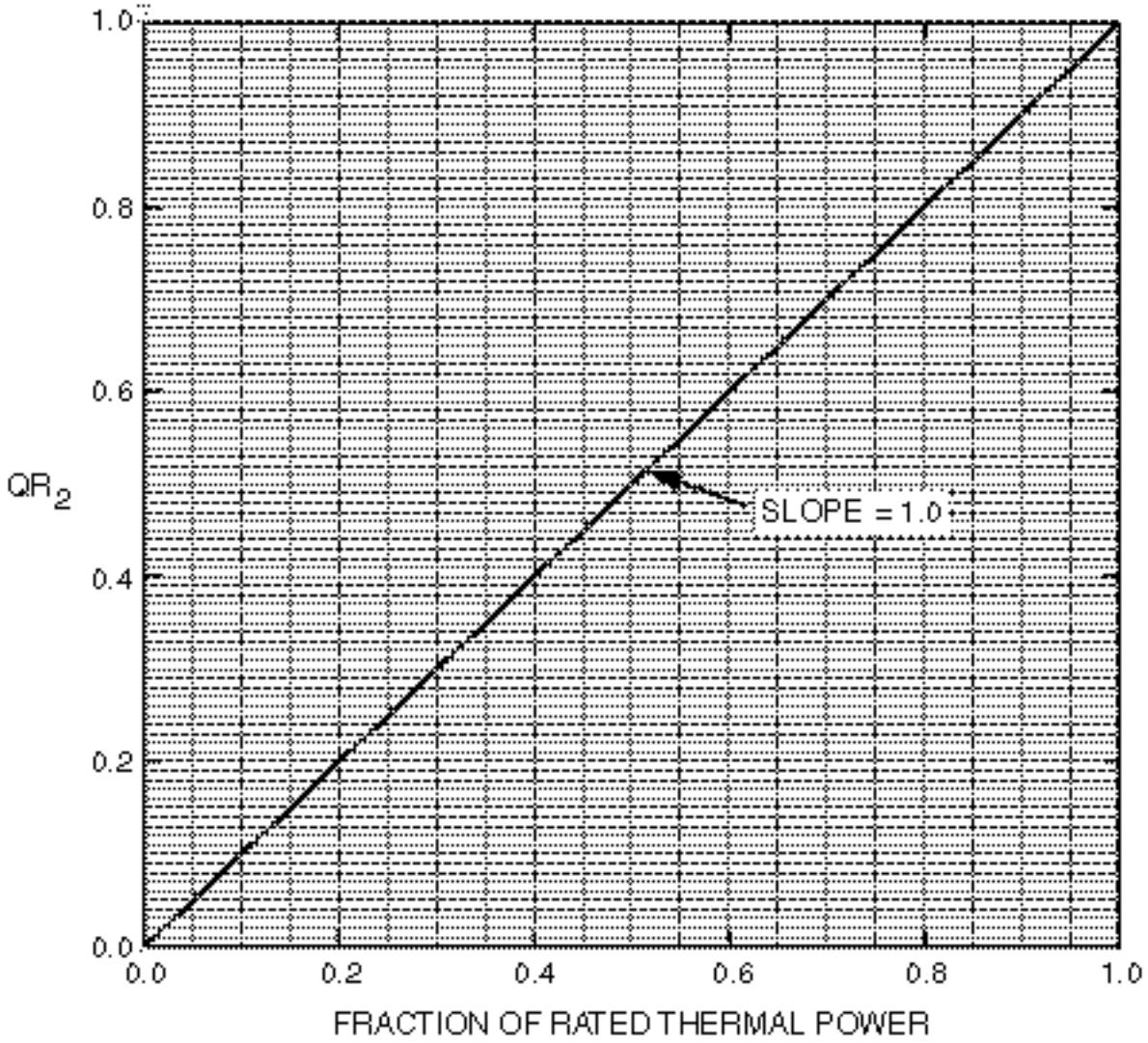

FIGURE 2.2-1

Local power density – High trip setpoint
Part 1 (Fraction of RATED THERMAL POWER versus QR_2)

See ITS 3.3.1

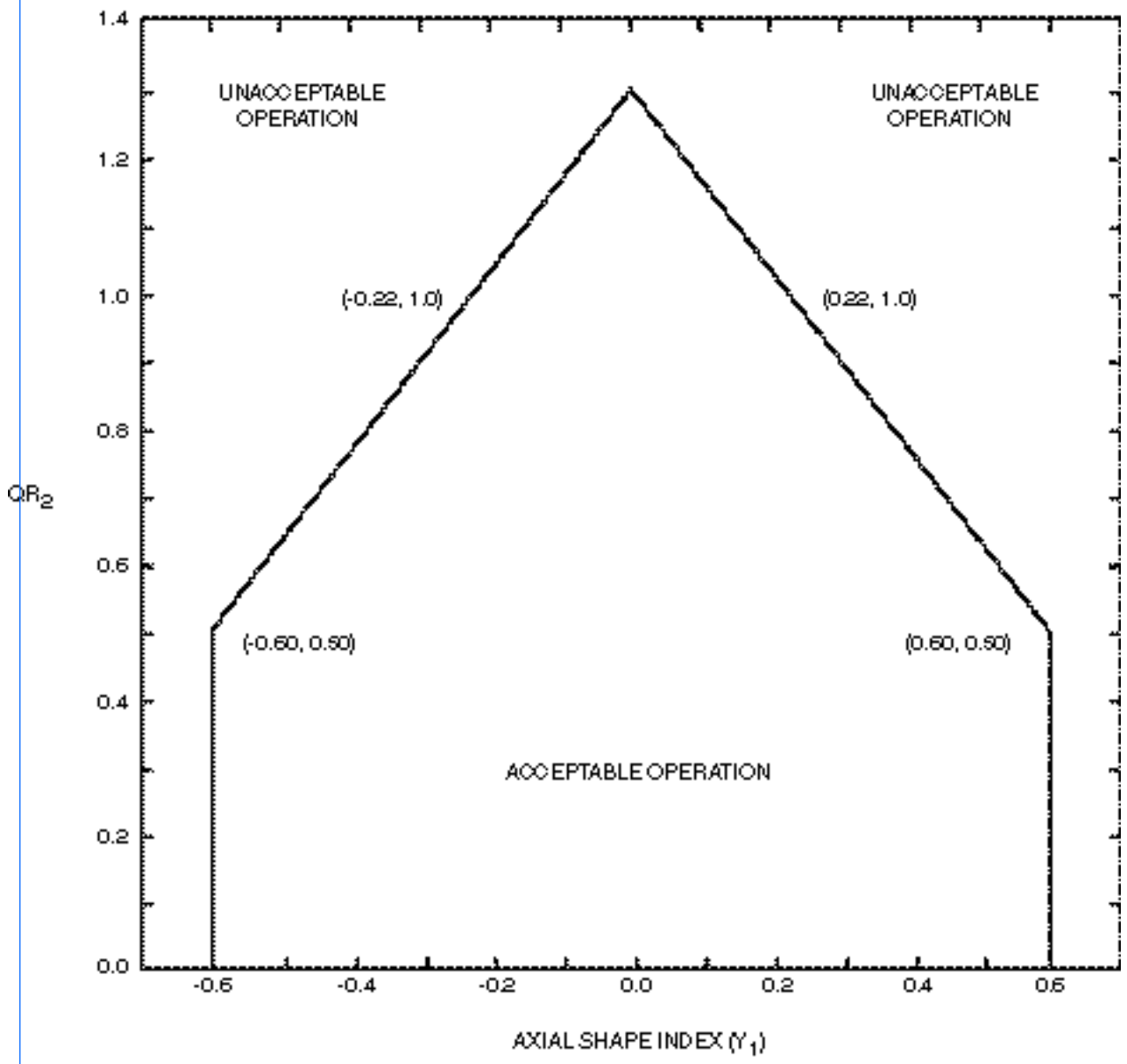

FIGURE 2.2-2

LOCAL POWER DENSITY – HIGH TRIP SETPOINT PART 2
(QR_2 versus Y_1)

See ITS 3.3.1

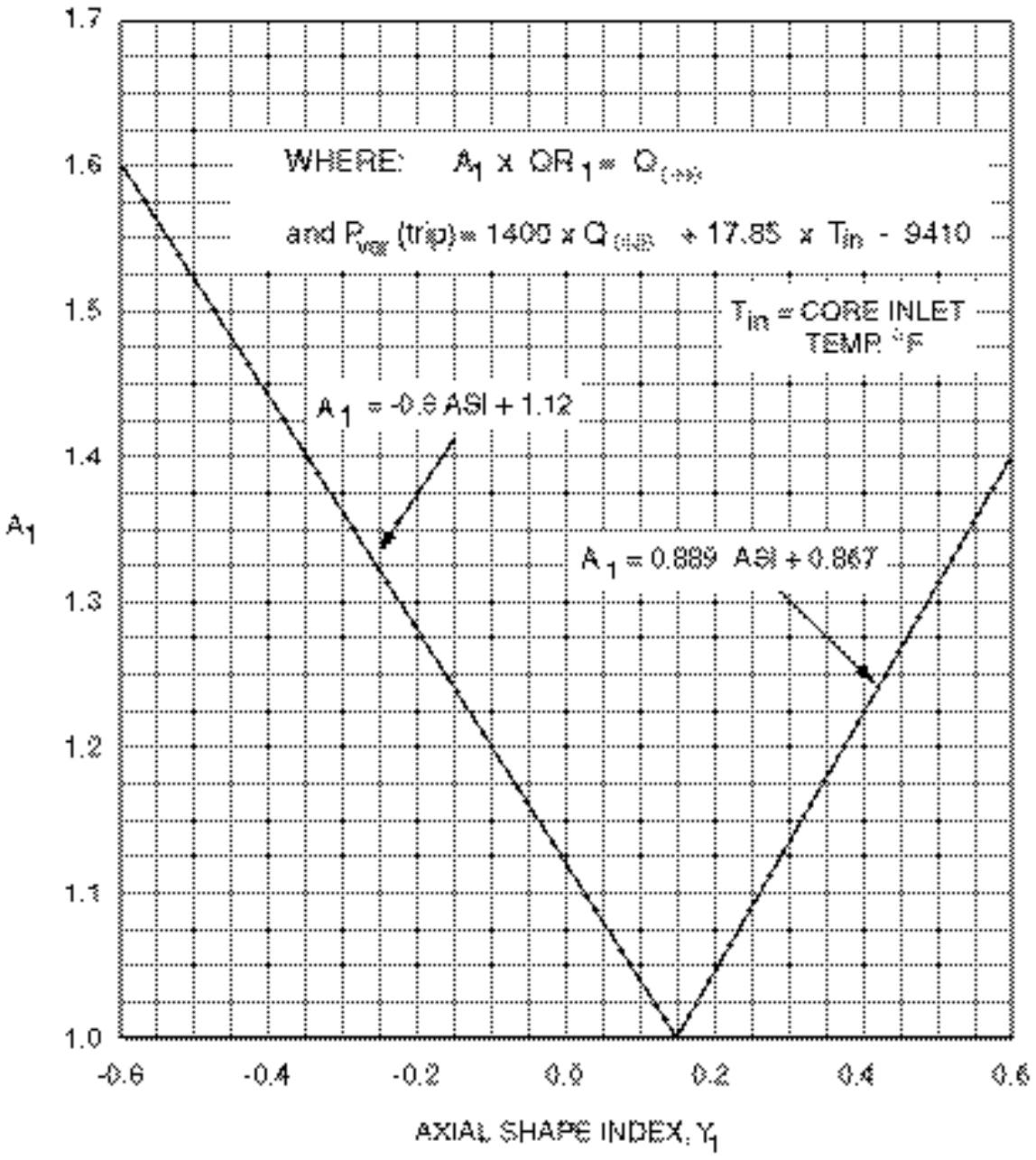

FIGURE 2.2-3

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 1 (Y_1 Versus A_1)

See ITS 3.3.1

WHERE: $A_1 \times QR_1 = Q_{DNB}$

AND $p_{var}^{trip} = 1400 \times Q_{DNB} + 17.85 \times T_{in} - 9410$

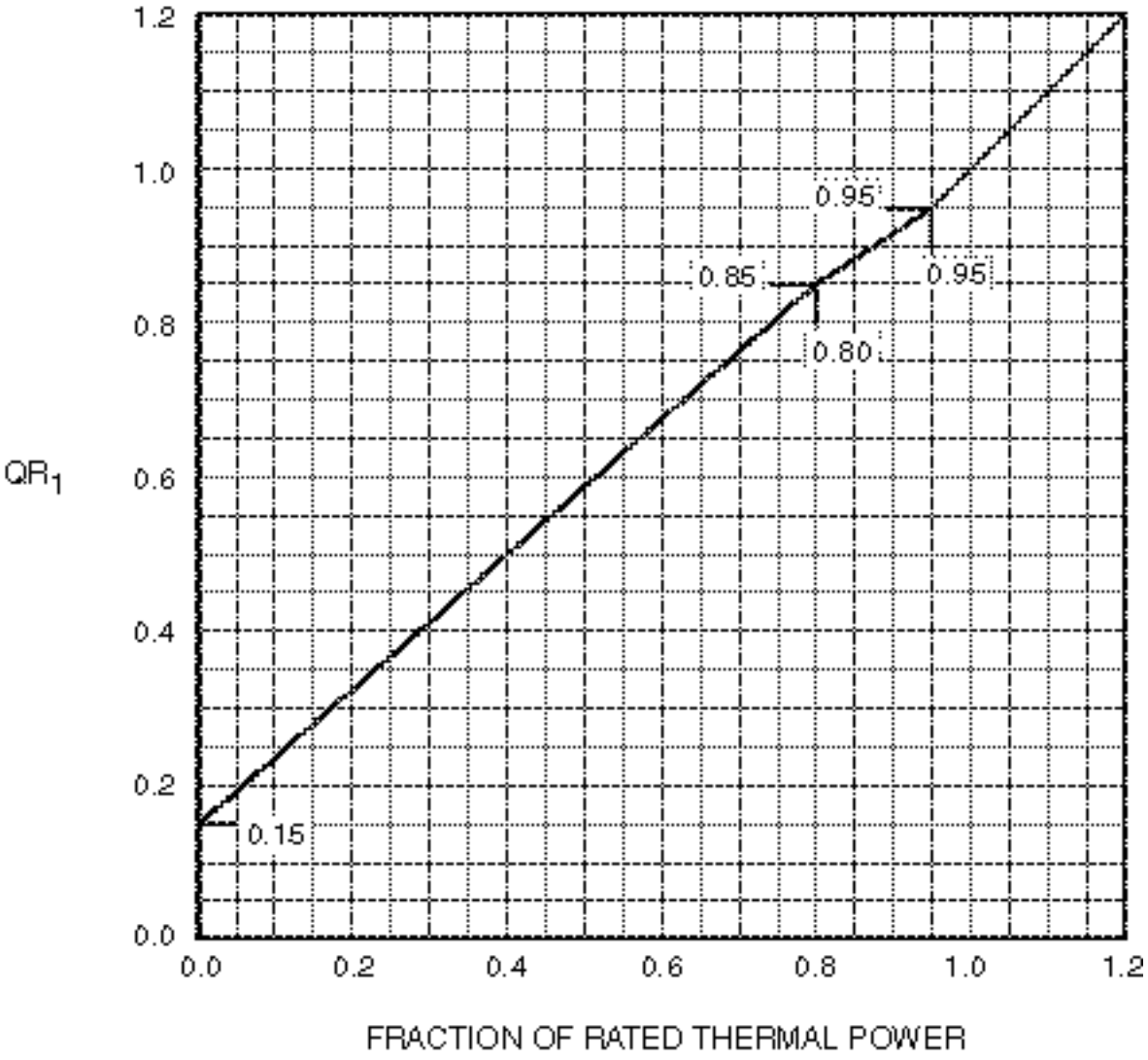

FIGURE 2.2-4

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 2 (FRACTION OF RATED THERMAL POWER VERSUS QR_1)

DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 1 only:** CTS Figure 2.1-1 states “Reactor Coolant System total flow rate – specified in LCO 3.2.5.” ITS Figure 2.1.1-1 does not contain the statement. This changes the CTS by not including a cross reference to the equivalent reference ITS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," which contains the Reactor Coolant System total flowrate. This change is acceptable because the CTS Figure 2.1-1 detail used to alert the user that the Reactor Coolant System total flowrate is covered under a separate LCO requirement is an unnecessary presentation preference. ITS does not include information only type of cross references to other Specifications. ITS Figure 2.1.1-1, Reactor Core Thermal Margin Safety Limit, is retained without the cross reference detail.

This change is designated as administrative as it incorporates an ITS convention with no technical change.

A03 PSL Unit 1 CTS Figure 2.1-1, Reactor Core Thermal Margin Safety Limit – Four Reactor Coolant Pumps Operating contains an operational limit line showing thermal power limited to a maximum of 112% of RATED THERMAL POWER by the High Power Trip. PSL Unit 2 CTS Figure 2.1-1, Reactor Core Thermal Margin Safety Limit – Four Reactor Coolant Pumps Operating contains an operational limit line showing thermal power limited to a maximum of 107% of RATED THERMAL POWER by the High Power Level Trip, and contains an operational limit line showing limits on maximum cold leg temperature due to secondary safety valve actuation. Additionally, the Unit 1 and Unit 2 figure titles include the statement “Four Reactor Coolant Pumps Operating.”

The RPS operational limit line is removed from the Unit 1 and Unit 2 figures. This operational limit could be violated during a transient; however, it does not necessarily mean the reactor core SL was violated. Removing the operational limit line shifts the unacceptable area to the actual SL limits based on RCS pressure lines. The RPS operational limit is provided in the ITS Bases. The figure title text “Four Reactor Coolant Pumps Operating” is deleted from the Unit 1 and Unit 2 figures. Unit 1 and Unit 2 only operate with four reactor coolant pumps. Therefore, this text is deleted for consistency with ISTS Figure 2.1.1-1 title. The operational limit line showing limits on maximum cold leg temperature due to secondary safety valve actuation is removed from the Unit 2 figure. This operational limit could be violated during a transient; however, it does not necessarily mean the reactor core SL was violated. Removing this operational

limit line shifts the unacceptable area to the actual SL limits based on RCS pressure lines.

This change is designated as administrative as it incorporates an ITS convention to remove superfluous lines not directly related to violation of the safety limit with no technical change.

MORE RESTRICTIVE CHANGES

M01 CTS 2.1.1 does not contain a Safety Limit to ensure fuel centerline temperature remains below the fuel melt temperature during normal operating conditions or design anticipated operational occurrences with adjustments for burnup and burnable poison. ITS requires Reactor Core Safety Limit 2.1.1.2, to ensure fuel centerline temperature remains below the fuel melt temperature during normal operating conditions or design anticipated operational occurrences with adjustments for burnup and burnable poison.

The purpose of ITS 2.1.1.2 is to provide a Safety Limit to ensure fuel centerline temperature remains below the fuel melt temperature during normal operating conditions or design anticipated operational occurrences with adjustments for burnup and burnable poison. This change is acceptable because it adds a Reactor Core Safety Limit that is not in the CTS.

This change is designated as more restrictive because it adds a Reactor Core Safety Limit to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 2.1.1 states “The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.” ITS states “The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figure 2.1.1-1.” This changes the CTS by moving details shown on the CTS Figure 2.1-1 to the ITS Bases.

PSL Unit 1 CTS Figure 2.1-1 statements “Reactor operation limited to less than 580°F by actuation of Main Steam Safety Valves”, “Limits contain no allowance for instruments error or fluctuations”, “Based on the Axial Shape Index on Figure B 2.1-1,” and “THERMAL POWER limited to a maximum of 112% of RATED THERMAL POWER by the high power trip level” are moved from the Technical Specifications to the ITS Bases. The line showing THERMAL POWER at 112%

is deleted. The Reactor Core Safety Limits are retained in Technical Specification Chapter 2.0.

PSL Unit 2 CTS Figure 2.1-1 statements “Reactor Conditions limited to less than the temperatures defined by the actuation of the secondary safety valves”, “Limits contain no allowance for instruments error or fluctuations”, “Valid for Total Integrated Radial Peaking Factors less than specified in Tech Spec 3.2.3” and “THERMAL POWER limited to a maximum of 107% of RATED THERMAL POWER by the high power level trip” are moved from the Technical Specifications to the ITS Bases. The Reactor Core Safety Limits are retained in Technical Specification Chapter 2.0.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 2.1.1.1 retains the Reactor Core Safety Limit and Figure 2.1.1-1, Reactor Core Thermal Margin Safety Limit. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

2.0 SAFETY LIMITS (SLs) ~~(Analog)~~

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figure 2.1.1-1.

2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained ~~at < [5080]°F, decreasing by [58°F per 10,000 MWD/MTU] and adjusted for burnable poison per [CENPD-275-P, Revision 1-P-A or CENPD-382-P-A]~~.

below fuel melt temperature limits specified in the COLR

2

2.1.2 Reactor Coolant System Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[~~2750~~]~~ psia.

2.2 SAFETY LIMIT VIOLATIONS

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3

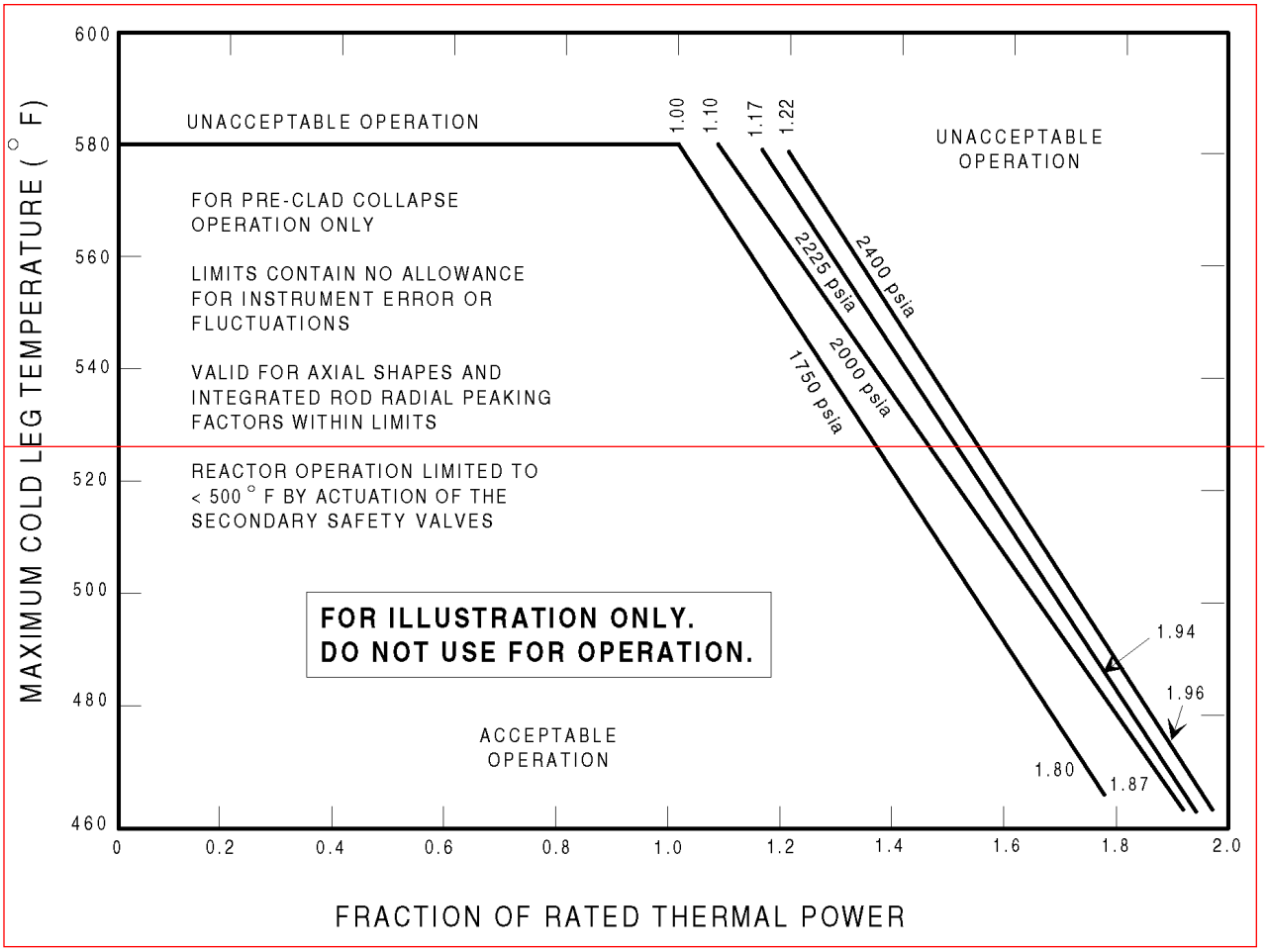

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Figure 2.1.1-1 (page 1 of 1)
Reactor Core Thermal Margin Safety Limit

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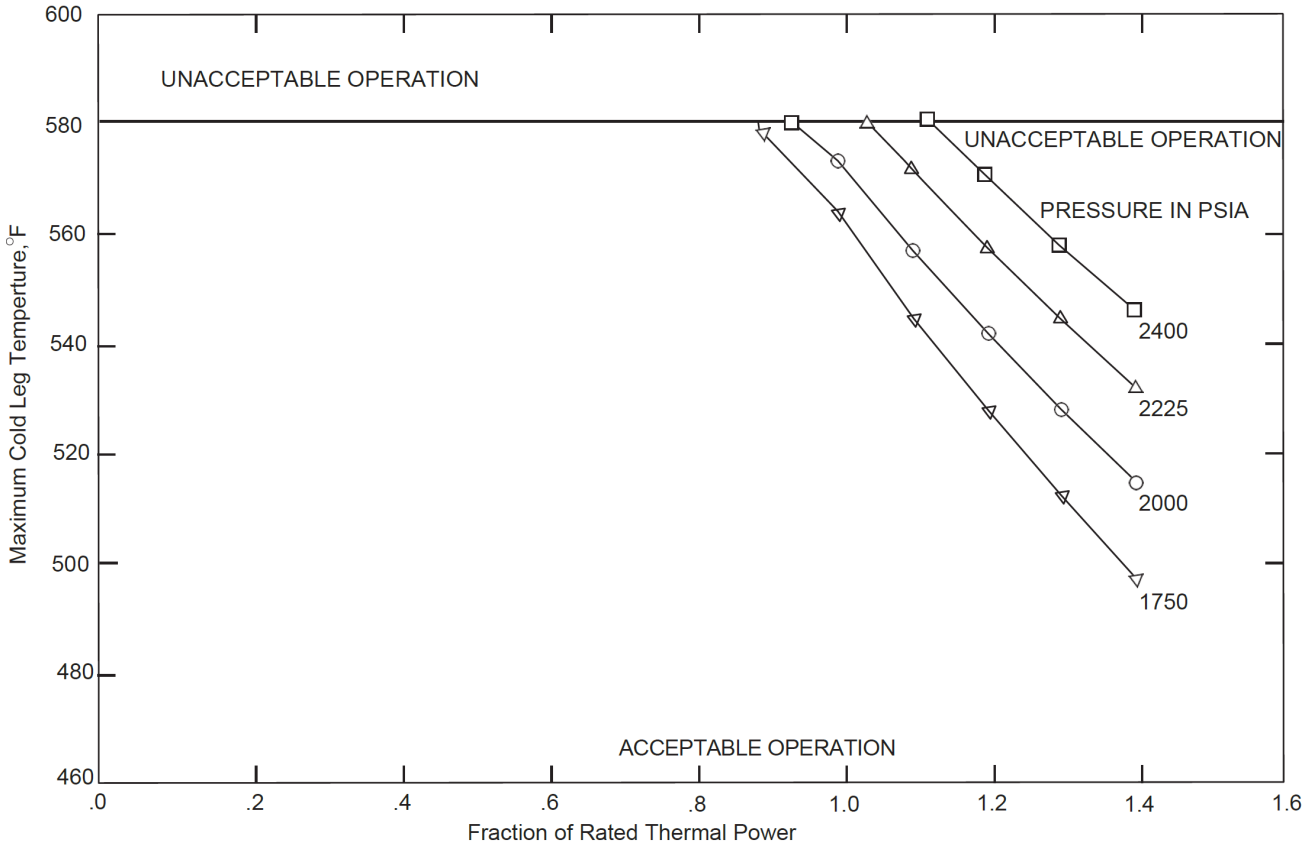
Unit 1 Only



2.0 SAFETY LIMITS (SLs) ~~(Analog)~~

2.1

2.1 SLs

2.1.1

2.1.1 Reactor Core SLs

2.1.1 and Applicability

2.1.1.1 In MODES 1 and 2, the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figure 2.1.1-1.

DOC M01

2.1.1.2 In MODES 1 and 2, peak fuel centerline temperature shall be maintained ~~at < [5080]°F, decreasing by [58°F per 10,000 MWD/MTU] and adjusted for burnable poison per [CENPD-275-P, Revision 1-P-A or CENPD-382-P-A]~~. 2

below fuel melt temperature limits specified in the COLR

2.1.2

2.1.2 Reactor Coolant System Pressure SL

2.1.2 and Applicability

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[~~2750~~]~~ psia.

2.2 SAFETY LIMIT VIOLATIONS

2.1.1 ACTION

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.1.2 ACTION

2.2.2 If SL 2.1.2 is violated:

2.1.2 ACTION

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.1.2 ACTION

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3

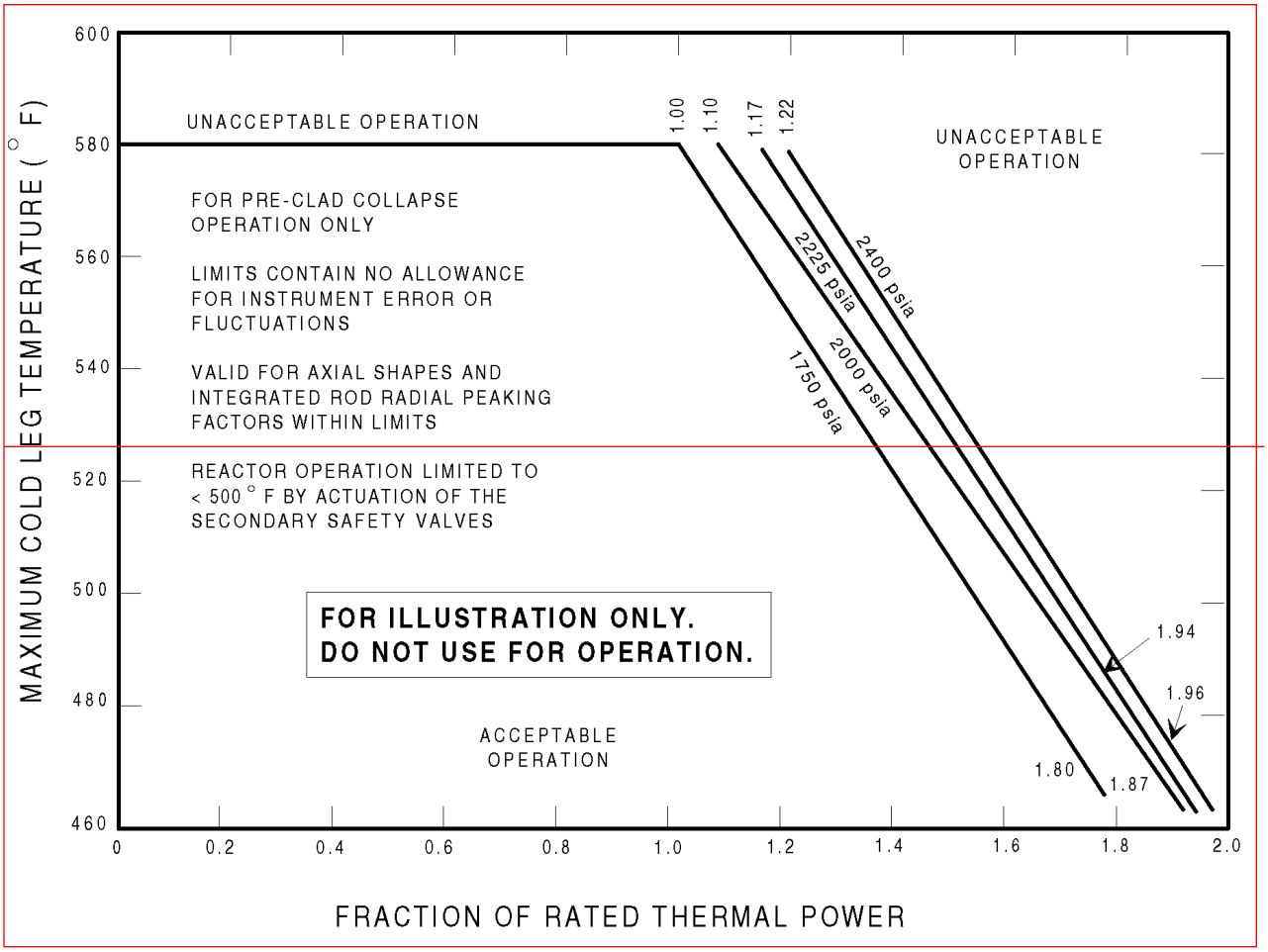


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Figure 2.1.1-1 (page 1 of 1)
Reactor Core Thermal Margin Safety Limit

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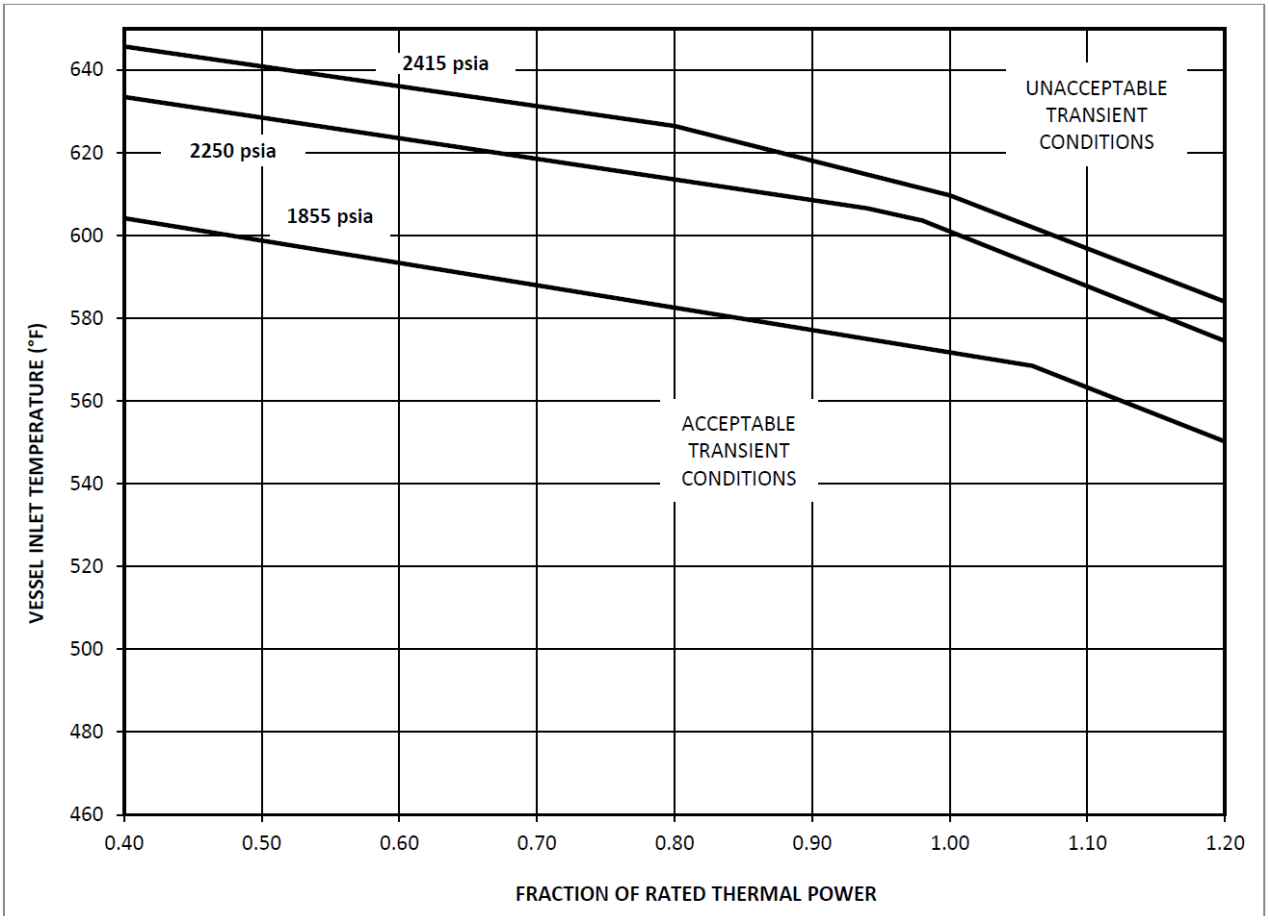
Unit 2 Only



JUSTIFICATION FOR DEVIATIONS
ISTS CHAPTER 2.0, SAFETY LIMITS (SLs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis. PSL does not currently require an SL regarding peak fuel centerline temperature. However, peak fuel centerline temperature is maintained below the fuel melt temperature specified in the COLR. The fuel melt temperature is determined using the NRC approved methodology in Siemens Topical Report XN-NF-79-56(P)(A) Revision 1 and Supplement 1, “Gadolinia Fuel Properties for LWR Fuel Safety Evaluation,” dated October 1981.

3. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 2.0 SAFETY LIMITS (SLs) ~~(Analog)~~

B 2.1.1 Reactor Core SLs ~~(Analog)~~ 5

BASES

BACKGROUND

GDC 10 (Ref. 1) requires and SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operation transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The Reactor Protective System (RPS), in combination with the LCOs, is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints, LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for RCS temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. Pressurizer Pressure High trip,

b. Variable ~~High Power~~ trip, (1) R1
Power Level - High

c. Power Rate of Change - High trip,

d. Reactor Coolant Flow - Low trip,

e. Steam Generator Pressure - Low trip,

f. Steam Generator Level - Low trip,

g. ~~Axial Power Distribution~~ - High trip, (1)
Local Power Density

h. Thermal Margin/Low Pressure trip,

i. Steam Generator Pressure Difference trip, and

j. Steam Generator Safety Valves.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.~~5~~, "AXIAL SHAPE INDEX (ASI)," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. (1)
4 U

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BASES

The SL contains no allowance for instrument error. Actuation of the main steam safety valves limits reactor coolant temperature.

4

SAFETY LIMITS

The curves provided in Figure ~~B~~ 2.1.1-1 show the loci of points of THERMAL POWER, pressurizer pressure, and highest operating loop cold leg temperature, for which the minimum ~~-~~DNBR is not less than the safety analysis limit. SL 2.1.1.2 ensures that fuel centerline temperature remains below melting.

3

SL 2.1.1.2 ensures that fuel centerline temperature remains below the fuel melt temperature ~~of [5080]°F~~ during normal operating conditions or design AOOs with adjustments for burnup and burnable poison. ~~An adjustment of [58°F per 10,000 MWD/MTU] has been established in [Topical Report CEN-386-P-A] (Ref. 3) and adjustments for burnable poisons are established based on [Topical Reports CENPD-275-P] (Ref. 4) and [CENPD-382-P-A] (Ref.5).~~

limits specified in the COLR

2

2

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip ~~-~~function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

3

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable and reduces the probability of fuel damage.

Fuel melt temperature is determined using the NRC approved methodology in Reference 3. The fuel melt temperature limit is defined as [(2790 - 17.9 x P - 3.2 x B) x 1.8 + 32] °F, where P is the maximum weight percent of Gadolinia (%) and B is the maximum pin burnup (GWD/MTU).

2

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BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988.

2. FSAR, Section ~~[~~7.2 ~~]~~. U 2

~~3. [Topical Report CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," August 1992.]~~ 2

~~4. [Topical Report CENPD-275-P, Revision 1-P-A, "CE Methodology for Core Designs Containing Gadolini-Urania Burnable Absorbers," May 1988.]~~ 2

~~5. [Topical Report CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.]~~ 2

3. Siemens Power Corporation Topical Report XN-NF-79-56(P)(A) Revision 1 and Supplement 1, "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," October 1981. 2

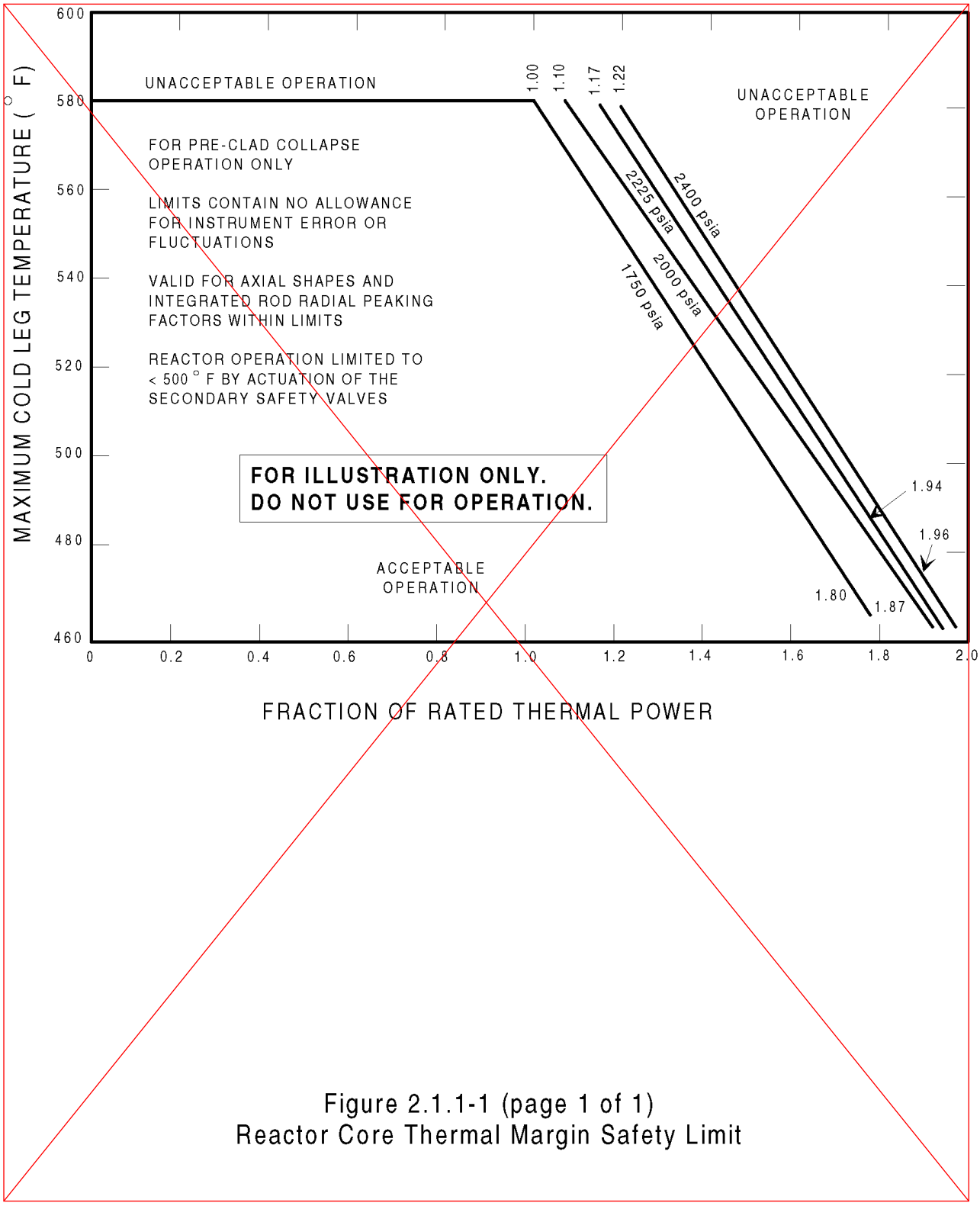

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Thermal Margin Safety Limit

5

B 2.0 SAFETY LIMITS (SLs) ~~(Analog)~~

5

B 2.1.2 Reactor Coolant System (RCS) Pressure SL ~~(Analog)~~

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, continued RCS integrity is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, according to GDC 28 (Ref. 1), "Reactivity Limits," reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, the RCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation, when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If this occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the Reactor Pressure - High trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and

BASES

APPLICABLE SAFETY ANALYSES (continued)

turbine-generator

hence the valve size requirements and lift settings, is a ~~[~~complete loss of ~~external~~ load without a direct reactor trip~~]~~. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained. 2

The Reactor Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the pressurizer safety valves, provide pressure protection for normal operation and AOOs. In particular, the Pressurizer Pressure - High trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Safety analyses for both the Pressure - High trip and the RCS pressurizer safety valves are performed, using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs),

b. Steam Bypass Control System,

c. Pressurizer Level Control System, or

d. Pressurizer Pressure Control System.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings under ~~[~~USAS, Section B31.1 (Ref. 6)~~]~~, is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 2750 psia. 2

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

5

BASES

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

2.2.2.1

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

With RCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater than the value specified in SL 2.1.2 exceeds 110% of the RCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III~~, Article NB-7000~~.

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3. ASME, Boiler and Pressure Vessel Code, Section XI~~, Article IWX-5000~~.
4. 10 CFR 100.
5. FSAR, Section ~~[~~15.2.7~~]~~.

2

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~~[~~ 6. ASME, USAS B31.1, Standard Code for Pressure Piping, 1967. ~~]~~

2

B 2.0 SAFETY LIMITS (SLs) ~~(Analog)~~ 5

B 2.1.1 Reactor Core SLs ~~(Analog)~~

BASES

BACKGROUND

GDC 10 (Ref. 1) requires and SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operation transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The Reactor Protective System (RPS), in combination with the LCOs, is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints, LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for RCS temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. Pressurizer Pressure High trip,

b. Variable ~~High Power~~ trip, (1) R1
Power Level - High

c. Power Rate of Change - High trip,

d. Reactor Coolant Flow - Low trip,

e. Steam Generator Pressure - Low trip,

f. Steam Generator Level - Low trip,

g. ~~Axial Power Distribution~~ - High trip, (1)
Local Power Density

h. Thermal Margin/Low Pressure trip,

i. Steam Generator Pressure Difference trip, and

j. Steam Generator Safety Valves.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.2.1, "Linear Heat Rate (LHR) ," and LCO 3.2.~~5~~, "AXIAL SHAPE INDEX (ASI)," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. (1)
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BASES

The SL contains no allowance for instrument error. Actuation of the main steam safety valves limits reactor coolant temperature.

4

SAFETY LIMITS

The curves provided in Figure ~~B~~ 2.1.1-1 show the loci of points of THERMAL POWER, pressurizer pressure, and highest operating loop cold leg temperature, for which the minimum ~~-~~DNBR is not less than the safety analysis limit. SL 2.1.1.2 ensures that fuel centerline temperature remains below melting.

3

SL 2.1.1.2 ensures that fuel centerline temperature remains below the fuel melt temperature ~~of [5080]°F~~ during normal operating conditions or design AOOs with adjustments for burnup and burnable poison. ~~An adjustment of [58°F per 10,000 MWD/MTU] has been established in [Topical Report CEN-386-P-A] (Ref. 3) and adjustments for burnable poisons are established based on [Topical Reports CENPD-275-P] (Ref. 4) and [CENPD-382-P-A] (Ref.5)~~.

limits specified in the COLR

2

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APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip ~~-~~function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

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In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable and reduces the probability of fuel damage.

Fuel melt temperature is determined using the NRC approved methodology in Reference 3. The fuel melt temperature limit is defined as [(2790 - 17.9 x P - 3.2 x B) x 1.8 + 32] °F, where P is the maximum weight percent of Gadolinia (%) and B is the maximum pin burnup (GWD/MTU).

2

1

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988.

2. FSAR, Section ~~[~~7.2 ~~]~~. 2

U

~~3. [Topical Report CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," August 1992.]~~ 2

~~4. [Topical Report CENPD-275-P, Revision 1-P-A, "CE Methodology for Core Designs Containing Gadolini-Urania Burnable Absorbers," May 1988.]~~ 2

~~5. [Topical Report CENPD-382-P- A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.]~~ 2

3. Siemens Power Corporation Topical Report XN-NF-79-56(P)(A) Revision 1 and Supplement 1, "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," October 1981. 2

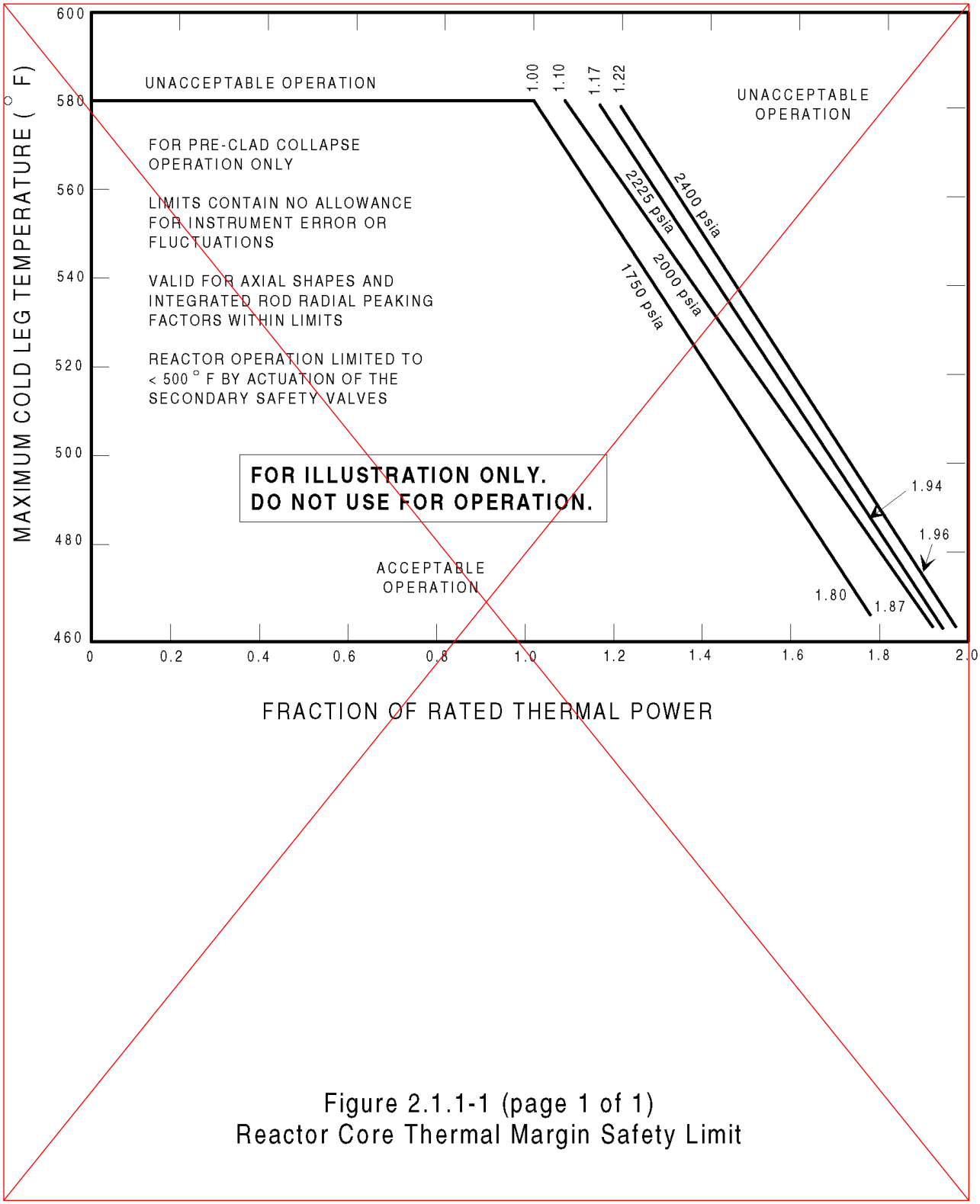

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Thermal Margin Safety Limit

B 2.0 SAFETY LIMITS (SLs) ~~(Analog)~~ 5

B 2.1.2 Reactor Coolant System (RCS) Pressure SL ~~(Analog)~~

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, continued RCS integrity is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, according to GDC 28 (Ref. 1), "Reactivity Limits," reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, the RCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation, when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If this occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the Reactor Pressure - High trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and

BASES

APPLICABLE SAFETY ANALYSES (continued)

turbine-generator

hence the valve size requirements and lift settings, is a ~~[~~complete loss of ~~external~~ load without a direct reactor trip~~]~~. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained. 2

The Reactor Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the pressurizer safety valves, provide pressure protection for normal operation and AOOs. In particular, the Pressurizer Pressure - High trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Safety analyses for both the Pressure - High trip and the RCS pressurizer safety valves are performed, using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs),

b. Steam Bypass Control System,

c. Pressurizer Level Control System, or

d. Pressurizer Pressure Control System.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings under ~~[~~USAS, Section B31.1 (Ref. 6)~~]~~, is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 2750 psia. 2

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

5

BASES

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

2.2.2.1

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

With RCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater than the value specified in SL 2.1.2 exceeds 110% of the RCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce RCS pressure by terminating the cause of the pressure increase, removing mass or energy from the RCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

2. ASME, Boiler and Pressure Vessel Code, Section III~~, Article NB-7000~~.

1

3. ASME, Boiler and Pressure Vessel Code, Section XI~~,~~ ~~Article IWX-5000~~.

4. 10 CFR 100.

5. FSAR, Section ~~[~~15.2.7~~]~~.

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~~[~~ 6. ASME, USAS B31.1, Standard Code for Pressure Piping, 1967. ~~]~~

2

JUSTIFICATION FOR DEVIATIONS
ISTS CHAPTER 2.0, BASES, SAFETY LIMITS (SLs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Grammatical error is corrected. Double spacing between sentences and single spacing between words. Single spacing is corrected.

4. CTS Figure 2.1-1 contained details that are moved to the Bases. The details are added to the ITS Figure 2.1.1-1 Bases. See DOC LA01.

5. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 2.0, SAFETY LIMITS (SLs)

There are no specific No Significant Hazards Considerations for this Specification.

ENCLOSURE 2

VOLUME 5

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.0
LCO AND SR APPLICABILITY

Revision 1

LIST OF ATTACHMENTS

1. ITS Section 3.0, LCO and SR Applicability

ATTACHMENT 1

ITS Section 3.0, LCO AND SR APPLICABILITY

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

LCO 3.0.1 3.0.1 Limiting Conditions for Operation (LCOs) shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2.

LCO 3.0.2 3.0.2 Upon discovery of a failure to meet an LCO, the ACTIONS shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified time interval(s), completion of the Required ACTIONS is not required, unless otherwise stated.

LCO 3.0.3 3.0.3 When a Limiting Condition for Operation (LCO) is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which specification does not apply by placing it, as applicable in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the LCO. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODES 5 or 6.

LCO 3.0.4 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate (exceptions to this Specification are stated in the individual Specifications); or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.5 3.0.5 Equipment removed from service or declared inoperable to comply with ACTION(s) may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

INSERT 1

are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in A03 A01

INSERT 2 A05

in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

~~APPLICABILITY~~

LCO Applicability

~~LIMITING CONDITION FOR OPERATION~~ (continued)

A01

LCO 3.0.6

3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the ~~ACTIONS~~ associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification ~~6.8.4.s~~, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate ~~ACTIONS~~ of the LCO in which the loss of safety function exists are required to be entered. When a support system's ~~ACTION~~ directs a supported system to be declared inoperable or directs entry into ~~ACTIONS~~ for a supported system, the applicable ~~ACTIONS~~ shall be entered in accordance with LCO 3.0.2.

5.5.12

Required Action

A01

Conditions and Required Actions

Insert 3

A06

Insert 4

A07

M01

Insert 5

L01

INSERT 3 A06

LCO 3.0.7 Special test exception (STE) LCO 3.1.8, "Special Test Exceptions (STE) – MODES 1 and 2" allows specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

INSERT 4 A07

LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours. M01

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

INSERT 5

L01

LCO 3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

A01

~~APPLICABILITY~~ 3.0

(SR) APPLICABILITY

A01

SURVEILLANCE REQUIREMENT~~S~~

SR 3.0.1

4.0.1 S~~urveillance~~ R~~equirement~~s shall be ~~applicable~~ during the ~~OPERATIONAL~~ MODES or other ~~conditions~~ specified for individual ~~Limiting Conditions for Operation~~ unless otherwise stated in ~~an individual Surveillance Requirement~~. Failure to meet a Surveillance ~~Requirement~~, whether such failure is experienced during the performance of the Surveillance ~~Requirement~~ or between performances of the Surveillance ~~Requirement~~, shall be failure to meet the LCO. Failure to perform a Surveillance ~~Requirement~~ within the ~~allowed surveillance interval~~ shall be failure to meet the LCO except as provided in SR ~~4~~.0.3. Surveillance ~~Requirement~~s do not have to be performed on inoperable equipment or variables outside specified limits.

conditions in the Applicability

met

LCOs,

the SR

specified Frequency

3

A01

A08

SR 3.0.2

4.0.2 ~~Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.~~

Insert 6

M02

L02

SR 3.0.3

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the L~~imiting~~ C~~ondition for~~ O~~peration~~ not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

A01

Insert 7

L03

If the Surveillance is not performed within the delay period, the L~~imiting~~ C~~ondition for~~ O~~peration~~ must immediately be declared not met, and the applicable ~~ACTION~~(s) must be ~~taken~~.

entered

Condition

When the Surveillance is performed within the delay period and the Surveillance is not met, the L~~imiting~~ C~~ondition for~~ O~~peration~~ must immediately be declared not met, and the applicable ~~ACTION~~(s) must be ~~taken~~.

entered

Condition

A01

SR 3.0.4

4.0.4 Entry into a MODE or other specified condition in the Applicability of a ~~Limiting Condition for Operation (~~LCO~~)~~ shall only be made when the LCO's Surveillances have been met within their specified ~~f~~frequency, except as provided by S~~urveillance~~ R~~equirement~~ ~~4~~.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

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This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

~~4.0.5 Surveillance Requirements for inservice inspection of ASME Code Class 1, 2 and 3 components shall be applicable as follows:~~

~~a. Inservice inspection of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).~~

~~b. deleted~~

LA01

INSERT 6

A08

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply. L02

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. M02

Exceptions to this Specification are stated in the individual Specifications.

INSERT 7

L03

The delay period is only applicable when there is a reasonable expectation the Surveillance will be met when performed.

~~APPLICABILITY~~

S~~URVEILLANCE R~~~~EQUIREMENTS~~ (Continued)

Applicability

A01

~~4.0.5 (Continued)~~

~~c. deleted~~

~~d. Performance of the above inservice inspection activities shall be in addition to other specified Surveillance Requirements.~~

~~e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.~~

LA01

PLANT SYSTEMS

3/4.7.10 SNUBBERS

LIMITING CONDITION FOR OPERATION

R1

3.7.10 All safety related snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

See ITS 3.7 Relocated-Deleted Current Technical Specifications CTS 3/4.7.10

ACTION:

LCO 3.0.8.a With one or more safety related snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.10 Each snubber shall be demonstrated OPERABLE by performance of the Snubber Testing Program.

See ITS 3.7 Relocated-Deleted Current Technical Specifications CTS 3/4.7.10

R1

Page Removed from submittal

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

LCO 3.0.1 3.0.1 Limiting Conditions for Operation (LCOs) shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2.

LCO 3.0.2 3.0.2 Upon discovery of a failure to meet an LCO, the ACTIONS shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified time interval(s), completion of the ACTIONS is not required, unless otherwise stated.

LCO 3.0.3 3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour, action shall be initiated to place the unit in a MODE in which specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation.

Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

LCO 3.0.4 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate (exceptions to this Specification are stated in the individual Specifications); or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.5 3.0.5 Equipment removed from service or declared inoperable to comply with ACTION(s) may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

INSERT 1

are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in A03 A01

INSERT 2 A05

in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

~~APPLICABILITY~~

LCO Applicability

~~LIMITING CONDITION FOR OPERATION~~ (continued)

A01

LCO 3.0.6

3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the ~~ACTIONS~~ associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification ~~6.8.4.t~~ 5.5.12 "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate ~~ACTIONS~~ of the LCO in which the loss of safety function exists are required to be entered.

A01

When a support system's ~~ACTION~~ Required Action directs a supported system to be declared inoperable or directs entry into ~~ACTIONS~~ Conditions and Required Actions for a supported system, the applicable ~~ACTIONS~~ shall be entered in accordance with LCO 3.0.2.

Insert 3

A06

Insert 4

A07

M01

Insert 5

L01

INSERT 3

A06

LCO 3.0.7 Special test exception (STE) LCO 3.1.8, “Special Test Exceptions (STE) – MODES 1 and 2” allows specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

INSERT 4

A07

LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

M01

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

INSERT 5

L01

LCO 3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

~~APPLICABILITY~~ 3.0

(SR) APPLICABILITY

SURVEILLANCE REQUIREMENT~~S~~

A01

SR 3.0.1 4.0.1 S~~urveillance~~ R~~equirement~~s shall be ~~applicable~~ met during the ~~OPERATIONAL~~ MODES or other ~~conditions~~ conditions in the Applicability specified for individual ~~Limiting Conditions for Operation~~ unless otherwise stated in ~~an individual Surveillance Requirement~~ the SR. Failure to meet a Surveillance ~~Requirement~~, whether such failure is experienced during the performance of the Surveillance ~~Requirement~~ or between performances of the Surveillance ~~Requirement~~, shall be failure to meet the LCO. Failure to perform a Surveillance ~~Requirement~~ within the ~~allowed~~ specified Frequency ~~surveillance interval~~ shall be failure to meet the LCO except as provided in SR ~~4~~.0.3. Surveillance ~~Requirement~~s do not have to be performed on inoperable equipment or variables outside specified limits.

LCOs,

3

A01

A08

SR 3.0.2 4.0.2 ~~Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.~~

Insert 6

M02

L02

SR 3.0.3 4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the L~~imiting~~ C~~ondition for~~ O~~peration~~ not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

A01

Insert 7

L03

If the Surveillance is not performed within the delay period, the L~~imiting~~ C~~ondition for~~ O~~peration~~ must immediately be declared not met, and the applicable ~~ACTION~~(s) must be ~~taken~~.

entered

Condition

When the Surveillance is performed within the delay period and the Surveillance is not met, the L~~imiting~~ C~~ondition for~~ O~~peration~~ must immediately be declared not met, and the applicable ~~ACTION~~(s) must be ~~taken~~.

entered

Condition

n

A01

SR 3.0.4 4.0.4 Entry into a MODE or other specified condition in the Applicability of a ~~Limiting Condition for Operation (~~LCO~~)~~ shall only be made when the LCO's Surveillances have been met within their specified ~~f~~frequency, except as provided by S~~urveillance~~ R~~equirement~~ ~~4~~.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

F

3

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

~~4.0.5 Surveillance Requirements for inservice inspection of ASME Code Class 1, 2 and 3 components shall be applicable as follows:~~

~~a. Inservice inspection of ASME Code Class 1, 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).~~

~~b. deleted~~

~~c. deleted~~

LA01

INSERT 6 A08

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply. M02

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. L02

Exceptions to this Specification are stated in the individual Specifications.

INSERT 7 L03

The delay period is only applicable when there is a reasonable expectation the Surveillance will be met when performed.

~~APPLICABILITY~~

Applicability

S~~URVEILLANCE R~~~~EQUIREMENTS~~ (Continued)

A01

~~4.0.5 (Continued)~~

~~d. Performance of the above inservice inspection activities shall be in addition to other specified Surveillance Requirements.~~

~~e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.~~

LA01

PLANT SYSTEMS

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

R1

3.7.9 All safety-related snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

See ITS 3.7 Relocated-Deleted Current Technical Specifications CTS 3/4.7.9

ACTION:

LCO 3.0.8.a

With one or more safety related snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9. on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the Snubber Testing Program.

a. Exemption From Visual Inspection or Functional Tests

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date.

See ITS 3.7 Relocated-Deleted Current Technical Specifications CTS 3/4.7.9

R1

Page removed from submittal

DISCUSSION OF CHANGES
ITS 3.0, LCO AND SR APPLICABILITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.0.1 states “Limiting Conditions for Operation (LCO) shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2." ITS LCO 3.0.1 states, "LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9." This results in changes to the CTS.

- ITS LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9 are added exceptions in ITS LCO 3.0.1. ITS LCO 3.0.7 describes Test Exceptions LCOs, which are exceptions to other LCOs. ITS LCO 3.0.8 addresses snubber inoperabilities, which is also an exception to other LCOs. ITS LCO 3.0.9 addresses barrier inoperabilities which is also an exception to other LCOs. Changes resulting from the incorporation of ITS LCO 3.0.8 are discussed in DOCs A07 and M01. Changes resulting from the incorporation of ITS LCO 3.0.9 are discussed in DOC L01.

  This change is acceptable because adding the exceptions for LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9 prevent a conflict within the Applicability section. This addition is needed for consistency in the ITS requirements and does not change the intent or application of the Technical Specifications.

- Certain phrases are revised to be consistent with the equivalent phrase used in the ITS. Specifically, "OPERATIONAL MODES or other conditions specified therein" is changed to "MODES or other specified conditions in the Applicability" to be consistent with the ITS definition of MODE and the terminology used in the ITS.

  This change is acceptable because it results in no change in the intent or application of the Technical Specifications, but merely reflect editorial preferences used in the ITS.

A03 CTS 3.0.3, in part, is applicable "When a Limiting Condition for Operation (LCO) is not met, except as provided in the associated ACTION requirements." ITS LCO 3.0.3 expands those applicability requirements so that the requirement is applicable "When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS." This changes the CTS to add two new applicability conditions.

- ITS LCO 3.0.3 is applicable when the LCO is not met and there is no applicable ACTION to be taken.

  This change is acceptable because it is consistent with the current understanding and application of CTS 3.0.3.

- ITS LCO 3.0.3 is applicable when directed by the associated ACTIONS. The CTS and the ITS contain such requirements. Any technical changes related to directing LCO 3.0.3 entry in an ACTION will be discussed in the affected Technical Specifications.

  This change is acceptable because it is consistent with the current understanding and application of CTS 3.0.3.

These changes are designated as administrative because they do not result in any technical changes to the Technical Specifications.

A04 CTS 3.0.3, in part, states that within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in: Hot Standby within the next 6 hours, Hot Shutdown within the following 6 hours, and at least Cold Shutdown within the subsequent 24 hours. ITS LCO 3.0.3 states that action shall be initiated within 1 hour to place the unit, as applicable, in MODE 3 within 7 hours, MODE 4 within 13 hours, and MODE 5 within 37 hours. This changes the CTS by using the sum of the times (i.e., the ITS Completion Time of 37 hours to enter MODE 5 is the same as the sum of the CTS allowance of 1 hour, 6 hours, 6 hours, and 24 hours) instead of sequential times (i.e., each time is measured from the completion of the previous step). The stated times in CTS 3.0.3 and ITS 3.0.3 are listed below:

Mode	Title	CTS Time to Enter Mode	ITS Time to Enter Mode
--	(Current Mode)	1 hour to begin action	1 hour to begin action
3	Hot Standby	within the next 6 hours	7 hours
4	Hot Shutdown	within the following 6 hours	13 hours
5	Cold Shutdown	within the subsequent 24 hours	37 hours

The purpose of CTS 3.0.3 is to establish the shutdown requirements that must be implemented when an LCO is not met and the condition is not specifically addressed in the associated ACTION requirements. The delineated time limit allows the unit to be placed in a safe shutdown MODE when the plant cannot be maintained within the limits for safe operation. The time limit, specified in CTS 3.0.3 to reach the lower MODES of operation, permits the shutdown to proceed in a controlled manner that is well within the specified maximum cooldown rate. Furthermore, the time limit is within the cooldown capabilities of the plant

assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies. In the CTS, this is accomplished by allowing a total of 37 hours for the plant to be in Cold Shutdown when a shutdown is required during the MODE of Operation. In the absence of specific guidance within the CTS, current PSL practice if the unit is in a lower MODE of Operation and a CTS 3.0.3 shutdown is required, is to apply the time limit for reaching the lower MODE of operation (i.e., each time limit is measured from the time the previous MODE is reached). In the ITS, the time limits for ITS LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower mode of operation when an ITS LCO 3.0.3 shutdown is required, the time limit for reaching the next lower MODE applies (i.e., if the plant is in MODE 3, 13 hours is allowed to reach MODE 4). ITS 3.0 Bases gives a detailed discussion on the use of applying the allowed outage times when the unit is in a lower MODE when ITS LCO 3.0.3 is entered. This is further explained, with examples, in the discussion of Section 1.3, "Completion Times." This change is acceptable because ITS and CTS both allow 37 hours to reach MODE 5 from power operation. In addition, the CTS 3.0.3 statement " within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply" has been editorially reworded in ITS LCO 3.0.3 to "the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. ACTION shall be initiated within 1 hour to place the unit..." These changes are considered changes to the CTS presentation. These changes are designated as administrative as they apply rules of usage established by ITS without resulting in technical changes to the Technical Specifications.

A05 CTS 3.0.3 states "Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the LCO." ITS LCO 3.0.3 states "Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4."

This change is acceptable because the changes to CTS 3.0.3 are editorial. Both the CTS and ITS state that LCO 3.0.3 can be exited if the LCO which led to the entry into LCO 3.0.3 is met, or if one of the ACTIONS of that LCO is applicable. The CTS requirement also specifies that the time to complete the ACTIONS in the LCO is based on the initial failure to meet the LCO. Reentering the LCO after exiting LCO 3.0.3 does not reset the ACTION statement time requirements. This information is not explicitly stated in ITS LCO 3.0.3 but is true under the multiple condition entry concept of the ITS.

In addition, CTS 3.0.3 states “This specification is not applicable in MODES 5 or 6.” ITS LCO 3.0.3 states “LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.” CTS 3.0.3 and ITS LCO 3.0.3 require the unit to be placed only as low as COLD SHUTDOWN (MODE 5). Once the unit is in MODE 5, there are no further requirements. Thus, CTS 3.0.3 and ITS LCO 3.0.3 are effectively only applicable

in MODES 1, 2, 3, and 4, and the addition of the sentence merely reflects editorial preferences used in the ITS.

These changes are designated as administrative because there is no change in the intent or application of the CTS 3.0.3 requirements.

A06 ITS LCO 3.0.7 is added to the CTS. ITS LCO 3.0.7 states "Special Test Exception (STE) LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications."

This change is acceptable because the CTS contain test exception specifications that allow certain LCOs to not be met for the purpose of special tests and operations. However, the CTS does not contain the equivalent of ITS LCO 3.0.7. As a result, there could be confusion regarding which LCOs are applicable during special tests. LCO 3.0.7 was crafted to avoid that possible confusion. LCO 3.0.7 is consistent with the use and application of CTS test exception Specifications and does not provide any new restriction or allowance. This change is designated as administrative because it does not technically change the Technical Specifications.

A07 CTS Section 3.0 does not contain an allowance when snubbers are unable to perform their associated support function. Unit 1 CTS 3.7.10 and Unit 2 CTS 3.7.9 provide the action for inoperable safety related snubbers and requires replacement or restoration of the inoperable snubber(s) within 72 hours or declare the supported system inoperable. Unit 1 CTS 3.7.10 and Unit 2 CTS 3.7.9 are relocated to the Technical Requirements Manual as part of the ITS conversion. Consistent with Technical Specifications Task Force (TSTF) traveler TSTF- 372, Revision 4, “Addition of LCO 3.0.8, Inoperability of Snubbers,“ dated April 23, 2004 (ADAMS Accession No. ML041200567) allowances, the proposed change to CTS 3.0, "LCO Applicability" adds a new LCO 3.0.8 with actions for inoperable required snubbers incorporated. When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and either: a) the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated function within 72 hours; or b) the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours. At the end of the specified period (i.e., 12 hours or 72 hours) snubbers must be able to perform their associated function(s), or the affected support system LCO(s) shall be declared not met and the associated ACTIONS taken.

ITS LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable, provided only a single subsystem is affected. This 72 hour time is consistent with the CTS. The impact and technical analysis of this change is discussed in detail in TSTF-372, Revision 4.

The change, except as discussed in DOC M01 is designated as administrative because there is no change in the intent or application of the Unit 1 CTS 3.7.10 and Unit 2 CTS 3.7.9 requirements as justified in TSTF-372, Revision 4.

A08 CTS 4.0.2 states, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states, "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This results in several changes to the CTS.

- ITS SR 3.0.2 adds to the CTS "For Frequencies specified as "once," the above interval extension does not apply." This is described in DOC M02.

- ITS SR 3.0.2 adds to the CTS "If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance." This is covered by DOC L02.

- CTS 4.0.2 states, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states, in part, "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency." This change is being made to be consistent with the ITS terminology and to clarify the concept of the specified SR Frequency being met.

  This change is acceptable since it does not change the intent of the requirements.

- ITS SR 3.0.2 is more specific regarding the state of the Frequency by stating, "as measured from the previous performance or as measured from the time a specified condition of the Frequency is met." This direction is consistent with the current use and application of the Technical Specifications.

  This change is acceptable because the ITS intent is the same as the CTS requirement.

- ITS SR 3.0.2 adds to the CTS "Exceptions to this Specification are stated in the individual Specifications."

  This change is acceptable because it reflects practices used in the ITS that are not used in the CTS. Any changes to a Technical Specification, by inclusion of such an exception, will be addressed in the affected Technical Specification.

The changes, except as discussed in DOC M02 and DOC L02, are designated as administrative because they reflect presentation and usage rules of the ITS without making technical changes to the Technical Specifications.

MORE RESTRICTIVE CHANGES

M01 CTS Section 3.0 does not contain an allowance when snubbers are unable to perform their associated support function. Unit 1 CTS 3.7.10 and Unit 2 CTS 3.7.9 provide the action for inoperable safety related snubbers and requires replacement or restoration of the inoperable snubber(s) within 72 hours or declare the supported system inoperable. Unit 1 CTS 3.7.10 and Unit 2 CTS 3.7.9 are relocated to the Technical Requirements Manual as part of the ITS conversion. Consistent with TSTF-372, Revision 4 allowances, the proposed change to CTS 3.0, "LCO Applicability" adds a new LCO 3.0.8 with actions for inoperable required snubbers incorporated. When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and either: a) the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated function within 72 hours; or b) the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours. At the end of the specified period (i.e., 12 hours or 72 hours) snubbers must be able to perform their associated function(s), or the affected support system LCO(s) shall be declared not met and the associated ACTIONS taken.

ITS LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable when the snubbers are associated with more than one train or subsystem of a multiple train or subsystem supported system. The 12 hour time is appropriate based on the relative risk considered with respect to other plant maintenance activities and integration of the of the risk assessment and management aspect of the LCO into the plant’s existing 10 CFR 50.65(a)(4) process as provided in the technical analysis of this change in TSTF-372, Revision 4.

The proposed change, except as discussed in DOC A07, is designated as more restrictive because it imposes a new shorter specified period for restoration of snubbers that are associated with more than one train or subsystem than allowed in Unit 1 CTS 3.7.10 and Unit 2 CTS 3.7.9.

M02 CTS 4.0.2 states, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states, "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This changes the CTS by adding "For Frequencies specified as "once," the above interval extension does not apply." The remaining changes to CTS 4.0.2 are discussed in DOCs A08 and L02.

The purpose of the 1.25 extension allowance to Surveillance Frequencies is to allow for flexibility in scheduling tests. This change is acceptable because Frequencies specified as "once" are typically condition-based Surveillances in which the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Frequency without extension in order to avoid operation in unacceptable conditions. This change is designated as more restrictive because an allowance to extend Frequencies by 25% is eliminated for some Surveillances.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 4.0.5 provides requirements for the inservice inspection of ASME Code Class 1, 2 and 3 components. The ITS does not include Inservice Inspection Program requirements. This changes the CTS by moving these Surveillance Requirements from the Technical Specifications to the Inservice Inspection (ISI) Program.

The removal of these requirements is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The Technical Specifications still retain requirements for the affected components to be OPERABLE. Also, this change is acceptable because these requirements will be adequately controlled by the ISI Program, which is required by 10 CFR 50.55a. Compliance with 10 CFR 50.55a is required by the operating licenses of Units 1 and 2. This change is designated as a less restrictive removal of requirement change because requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS Section 3.0 does not contain an allowance when barriers cannot support their support function. The proposed change to CTS 3.0, "LCO Applicability" adds a new LCO 3.0.9. The addition of LCO 3.0.9 to the CTS is to address barriers which cannot perform their related support function for Technical Specification systems. ITS LCO 3.0.9 allows barriers to be able to not perform their safety function for up to 30 days before declaring the supported system inoperable. Furthermore, due to this addition, an allowance is also needed in LCO 3.0.1. This allowance has been added. | R1

Barriers are defined as doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, which are designed to provide for the performance of the safety function for the Technical Specification system after the occurrence of one or more initiating events.

The barrier which cannot perform its related support function will be evaluated and managed under the Maintenance Rule plant configuration control requirement, 10 CFR 50.65(a)(4), and the associated industry guidance (NUMARC 93-01, Revision 4A). This provision is applicable whether the barrier is affected due to planned maintenance or due to a discovered condition. Should the risk assessment and risk management actions for a specific plant configuration or emergent condition not support the 30 day allowed time, the Maintenance Rule risk management determined allowed time and actions must be implemented or the supported system’s LCO be considered not met.

Application of LCO 3.0.9 is dependent on the OPERABILITY of at least one train or subsystem of the supported Technical Specification system and the system's ability to mitigate the consequences of the specified initiating events. However, during the 30 day period allowed by LCO 3.0.9, there exists the possibility that the train or subsystem required to be OPERABLE will unexpectedly become inoperable. Absent any further consideration, this would likely result in both trains of a Technical Specification required system being declared inoperable (i.e., the train supported by the barriers to which LCO 3.0.9 was being applied and the emergent condition of the inoperable train). This would likely result in entering LCO 3.0.3 and a rapid plant shutdown. While this scenario is of low likelihood, it is of very high consequence to PSL and, therefore, should be avoided unless necessary to avoid an actual plant risk. As a result, LCO 3.0.9 contains a provision which addresses the emergent condition of the required OPERABLE train or subsystem becoming inoperable while LCO 3.0.9 is being used. LCO 3.0.9 provides 24 hours to either restore the inoperable train or subsystem or to cease relying on the provisions of LCO 3.0.9 to consider the train or subsystem supported by the affected barrier(s) OPERABLE. This 24 hour period is not based on a generic risk evaluation, as it would be difficult to perform such an analysis in a generic fashion. Rather, plant risk during this 24 hour allowance is managed using the contemporaneous risk assessment and management required by 10 CFR 50.65(a)(4) and recognizes the unquantified advantage to plant safety of avoiding a plant shutdown with the associated transition risk.

A risk impact of the 30 day allowance for barriers was performed. All PSL initiating events are located on the table depicted in Technical Specifications Task Force (TSTF) traveler TSTF-427, Revision 2, “Allowance for Non Technical Specification Barrier Degradation on Supported System OPERABILITY,“ dated May 3, 2006 (ADAMS Accession No. ML061240055) or St. Lucie has evaluated the use of LCO 3.0.9 for a barrier protecting against an initiating event not on the table located in TSTF-427, Revision 2 and calculated the frequency ranges within the ranges in the table so the above analysis is applicable for those initiators. Therefore, LCO 3.0.9 can be utilized when inoperable barriers affect Systems, Structures, or Components.

L02 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.0.2 states, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states, " The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This changes the CTS by adding, " If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance." The remaining changes to CTS 4.0.2 are discussed in DOCs A08 and M02. | R1

This change is acceptable because the 25% Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to Required Actions that must be performed periodically. The initial performance is excluded because the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Completion Time with extension in order to avoid operation in unacceptable conditions. This change is designated as less restrictive because addition time is provided to perform some periodic Required Actions.

L03 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.0.3 states, in part, "If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition of Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance." ITS SR 3.0.3 states, in part, "If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition of Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed." This changes the CTS by adding, "The delay | R1

period is only applicable when there is a reasonable expectation the surveillance will be met when performed."

The purpose of CTS 4.0.3 is to establish flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been performed within the specified Frequency. One premise for this flexibility is that the SR was satisfactorily met in the past and the most probable result of performing the SR is the verification of conformance with the requirements. Therefore, there is a reasonable expectation the SR will be met when performed. However, there are instances in which an SR may not have been performed in the past, but there is still a reasonable expectation the SR will be met when performed. This change is acceptable because it supports the premise that there is a reasonable expectation that the SR will be met when performed. This change has been approved by the NRC in Technical Specifications Task Force (TSTF) traveler TSTF-529, Revision 4, “Clarify Use and Application Rules,“ dated April 21, 2016 (ADAMS Accession No. ML16060A455). The NRC staff determined the application of the delay period provided by ISTS SR 3.0.3 acceptable for use on SRs that have never been performed as long as licensees can provide an adequate determination of reasonable expectation the SR will be met when performed. This change is designated as less restrictive because an allowance to defer declaring affected equipment inoperable is extended to SRs that have not been performed if a reasonable expectation the surveillance will be met when performed can be shown.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

3.0.1 LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9.

3.0.2 LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

3.0.3 LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours,

b. ~~[~~MODE 4 within 13~~]~~ hours, and

c. MODE 5 within 37 hours.

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Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

3.0.4 LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate (exceptions to this Specification are stated in the individual Specifications); or

c. When an allowance is stated in the individual value, parameter, or other Specification.

3.0 LCO Applicability

3.0.4 LCO 3.0.4 (continued)

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0.5 LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

3.0.6 LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

DOC A06 LCO 3.0.7

Special test exception (STE) LCOs ~~[in each applicable LCO section]~~ allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

DOC A07 LCO 3.0.8

When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

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3.1.8; “Special Test Exceptions (STE) – MODES 1 and 2,”

3.0 LCO Applicability

DOC A07

LCO 3.0.8 (continued)

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

DOC L01

LCO 3.0.9

When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

4.0.1	SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
4.0.2	SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
4.0.3	SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. The delay period is only applicable when there is a reasonable expectation the Surveillance will be met when performed. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

3.0 SR Applicability

4.0.4

SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

3.0.1 LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9.

3.0.2 LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

3.0.3 LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours,

b. ~~[~~MODE 4 within 13~~]~~ hours, and

c. MODE 5 within 37 hours.

2

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

3.0.4 LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate (exceptions to this Specification are stated in the individual Specifications); or

c. When an allowance is stated in the individual value, parameter, or other Specification.

3.0 LCO Applicability

3.0.4 LCO 3.0.4 (continued)

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0.5 LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

3.0.6 LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

DOC A06 LCO 3.0.7

Special test exception (STE) LCOs ~~[in each applicable LCO section]~~ allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with STE LCOs is optional. When an STE LCO is desired to be met but is not met, the ACTIONS of the STE LCO shall be met. When an STE LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

DOC A07 LCO 3.0.8

When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

3.1.8; "Special Test Exceptions (STE) – MODES 1 and 2,"

3.0 LCO Applicability

DOC A07 LCO 3.0.8 (continued)

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

DOC L01 LCO 3.0.9

When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

4.0.1 **SR 3.0.1**

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 **SR 3.0.2**

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

4.0.3 **SR 3.0.3**

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. The delay period is only applicable when there is a reasonable expectation the Surveillance will be met when performed. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

3.0 SR Applicability

4.0.4

SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

JUSTIFICATION FOR DEVIATIONS
ITS 3.0, LCO AND SR APPLICABILITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is changed to reflect the current licensing basis.

3. Changes are made to be consistent with changes made to the referenced ISTS Specification.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.9 establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered, unless otherwise specified. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation. Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

BASES

LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The ACTIONS for not meeting a single LCO adequately manage any increase in plant risk, provided any unusual external conditions (e.g., severe weather, offsite power instability) are considered. In addition, the increased risk associated with simultaneous removal of multiple structures, systems, trains or components from service is assessed and managed in accordance with 10 CFR 50.65(a)(4). Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and either:

a. An associated Required Action and Completion Time is not met and no other Condition applies or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

BASES

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. Planned entry into LCO 3.0.3 should be avoided. If it is not practicable to avoid planned entry into LCO 3.0.3, plant risk should be assessed and managed in accordance with 10 CFR 50.65(a)(4), and the planned entry into LCO 3.0.3 should have less effect on plant safety than other practicable alternatives.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to enter lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met,

b. The LCO is no longer applicable,

c. A Condition exists for which the Required Actions have now been performed, or

d. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for entering the next lower MODE applies. If a lower MODE is entered in less time than allowed, however, the total allowable time to enter

BASES

LCO 3.0.3 (continued)

MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is entered in 2 hours, then the time allowed for entering MODE 4 is the next 11 hours, because the total time for entering MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to enter a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.16, "Fuel Storage Pool Water Level." LCO 3.7.16 has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.16 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.16 of "Suspend movement of irradiated fuel assemblies in fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

Spent
3
spent
3
3
spent
1

[The requirement to be in MODE 4 in 13 hours is plant specific and depends on the ability to cool the pressurizer and degas.]

2

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with either LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

BASES

LCO 3.0.4 (continued)

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered following entry into the MODE or other specified condition in the Applicability will permit continued operation within the MODE or other specified condition for an unlimited period of time. Compliance with ACTIONS that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made and the Required Actions followed after entry into the Applicability.

For example, LCO 3.0.4.a may be used when the Required Action to be entered states that an inoperable instrument channel must be placed in the trip condition within the Completion Time. Transition into a MODE or other specified in condition in the Applicability may be made in accordance with LCO 3.0.4 and the channel is subsequently placed in the tripped condition within the Completion Time, which begins when the Applicability is entered. If the instrument channel cannot be placed in the tripped condition and the subsequent default ACTION ("Required Action and associated Completion Time not met") allows the OPERABLE train to be placed in operation, use of LCO 3.0.4.a is acceptable because the subsequent ACTIONS to be entered following entry into the MODE include ACTIONS (place the OPERABLE train in operation) that permit safe plant operation for an unlimited period of time in the MODE or other specified condition to be entered.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk

BASES

LCO 3.0.4 (continued)

assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Regulatory Guide 1.160 endorses the guidance in Section 11 of NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

BASES

LCO 3.0.4 (continued)

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., RCS Specific Activity), and may be applied to other Specifications based on NRC plant specific approval.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

BASES

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate either:

a. The OPERABILITY of the equipment being returned to service or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance. LCO 3.0.5 should not be used in lieu of other practicable alternatives that comply with Required Actions and that do not require changing the MODE or other specified conditions in the Applicability in order to demonstrate equipment is OPERABLE. LCO 3.0.5 is not intended to be used repeatedly.

An example of demonstrating equipment is OPERABLE with the Required Actions not met is opening a manual valve that was closed to comply with Required Actions to isolate a flowpath with excessive Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) leakage in order to perform testing to demonstrate that RCS PIV leakage is now within limit.

Examples of demonstrating equipment OPERABILITY include instances in which it is necessary to take an inoperable channel or trip system out of a tripped condition that was directed by a Required Action, if there is no Required Action Note for this purpose. An example of verifying OPERABILITY of equipment removed from service is taking a tripped channel out of the tripped condition to permit the logic to function and indicate the appropriate response during performance of required testing on the inoperable channel. Examples of demonstrating the OPERABILITY of other equipment are taking an inoperable channel or trip system out of the tripped condition 1) to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system, or 2) to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

BASES

LCO 3.0.5 (continued)

The administrative controls in LCO 3.0.5 apply in all cases to systems or components in Chapter 3 of the Technical Specifications, as long as the testing could not be conducted while complying with the Required Actions. This includes the realignment or repositioning of redundant or alternate equipment or trains previously manipulated to comply with ACTIONS, as well as equipment removed from service or declared inoperable to comply with ACTIONS.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

BASES

LCO 3.0.6 (continued)

Specification 5.5.14, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

The following examples use Figure B 3.0-1 to illustrate loss of safety function conditions that may result when a TS support system is inoperable. In this figure, the fifteen systems that comprise Train A are independent and redundant to the fifteen systems that comprise Train B. To correctly use the figure to illustrate the SFDP provisions for a cross train check, the figure establishes a relationship between support and supported systems as follows: the figure shows System 1 as a support system for System 2 and System 3; System 2 as a support system for System 4 and System 5; and System 4 as a support system for System 8 and System 9. Specifically, a loss of safety function may exist when a support system is inoperable and:

a. A system redundant to system(s) supported by the inoperable support system is also inoperable (EXAMPLE B 3.0.6-1),

b. A system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2), or

c. A system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).

For the following examples, refer to Figure B 3.0-1.

EXAMPLE B 3.0.6-1

If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in Systems 5, 10, and 11.

EXAMPLE B 3.0.6-2

If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11.

BASES

LCO 3.0.6 (continued)

EXAMPLE B 3.0.6-3

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

If an evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

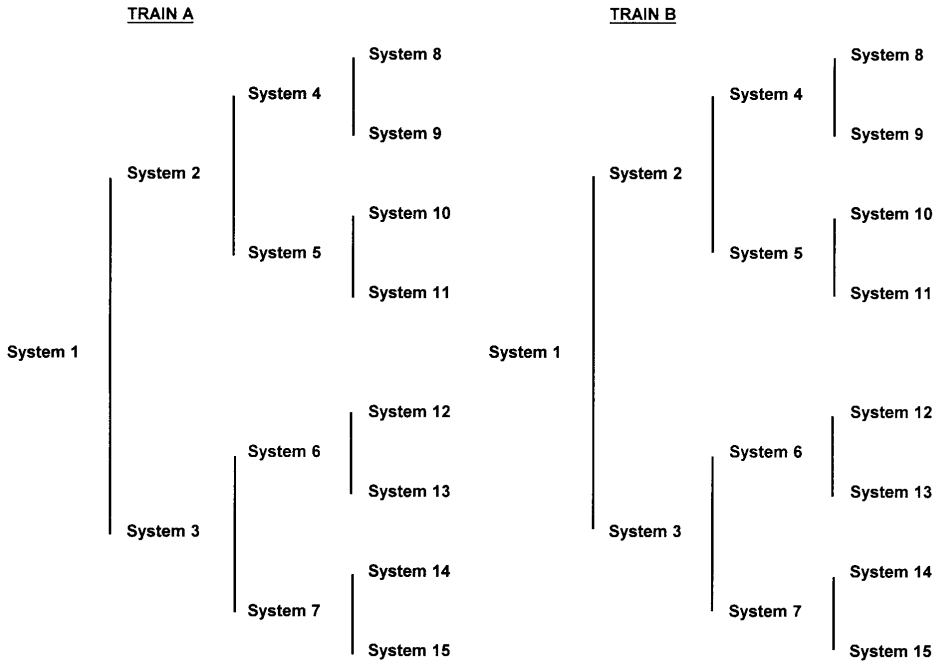

Figure B 3.0-1
Configuration of Trains and Systems

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC

BASES

LCO 3.0.6 (continued)

electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

Special tests and operations are required at various times over the unit's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, special test exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.

A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO

BASES

LCO 3.0.7 (continued)

apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.

Unless the SRs of the specified normal LCOs are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCOs must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

ACTIONS for STE LCOs provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system

BASES

LCO 3.0.8 (continued)

inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

LCO 3.0.9

~~REVIEWER'S NOTE~~

~~Adoption of LCO 3.0.9 requires the licensee to make the following commitments:~~

~~1. [LICENSEE] commits to the guidance of NUMARC 93-01, Revision [4F], Section 11, which provides guidance and details on the assessment and management of risk during maintenance.~~

~~2. [LICENSEE] commits to the guidance of NEI 04-08, "Allowance for Non Technical Specification Barrier Degradation on Supported System OPERABILITY (TSTF-427) Industry Implementation Guidance," March 2006.~~

LCO 3.0.9 establishes conditions under which systems described in the Technical Specifications are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

BASES

LCO 3.0.9 (continued)

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in the Technical Specifications. This LCO states that the supported system is not considered to be inoperable solely due to required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.9 allows 30 days before declaring the supported system(s) inoperable and the LCO(s) associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that as required barriers are found or are otherwise made unavailable, they are restored. However, the allowable duration may be less than the specified maximum time based on the risk assessment.

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers which are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001).

The provisions of LCO 3.0.9 are justified because of the low risk associated with required barriers not being capable of performing their related support function. This provision is based on consideration of the following initiating event categories:

~~REVIEWER'S NOTE~~

~~LCO 3.0.9 may be expanded to other initiating event categories provided plant-specific analysis demonstrates that the frequency of the additional initiating events is bounded by the generic analysis or if plant-specific approval is obtained from the NRC.~~

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- Loss of coolant accidents;
- High energy line breaks;
- Feedwater line breaks;
- Internal flooding;
- External flooding;
- Turbine missile ejection; and
- Tornado or high wind.

BASES

LCO 3.0.9 (continued)

The risk impact of the barriers which cannot perform their related support function(s) must be addressed pursuant to the risk assessment and management provision of the Maintenance Rule, 10 CFR 50.65 (a)(4), and the associated implementation guidance, Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Regulatory Guide 1.160 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This guidance provides for the consideration of dynamic plant configuration issues, emergent conditions, and other aspects pertinent to plant operation with the barriers unable to perform their related support function(s). These considerations may result in risk management and other compensatory actions being required during the period that barriers are unable to perform their related support function(s).

LCO 3.0.9 may be applied to one or more trains or subsystems of a system supported by barriers that cannot provide their related support function(s), provided that risk is assessed and managed (including consideration of the effects on Large Early Release and from external events). If applied concurrently to more than one train or subsystem of a multiple train or subsystem supported system, the barriers supporting each of these trains or subsystems must provide their related support function(s) for different categories of initiating events. For example, LCO 3.0.9 may be applied for up to 30 days for more than one train of a multiple train supported system if the affected barrier for one train protects against internal flooding and the affected barrier for the other train protects against tornado missiles. In this example, the affected barrier may be the same physical barrier but serve different protection functions for each train.

If during the time that LCO 3.0.9 is being used, the required OPERABLE train or subsystem becomes inoperable, it must be restored to OPERABLE status within 24 hours. Otherwise, the train(s) or subsystem(s) supported by barriers that cannot perform their related support function(s) must be declared inoperable and the associated LCOs declared not met. This 24 hour period provides time to respond to emergent conditions that would otherwise likely lead to entry into LCO 3.0.3 and a rapid plant shutdown, which is not justified given the low probability of an initiating event which would require the barrier(s) not capable of performing their related support function(s). During this 24 hour period, the plant risk associated with the existing conditions is assessed and managed in accordance with 10 CFR 50.65(a)(4).

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. SR 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 Specification.
SR 3.0.1	SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps. Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when either: a. The systems or components are known to be inoperable, although still meeting the SRs or b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances. Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a special test exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification. Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

BASES

SR 3.0.1 (continued)

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

b. High pressure safety injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

BASES

SR 3.0.2 (continued)

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.2 are applicable, a 25% extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted.

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs.

The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Examples of where SR 3.0.2 does not apply are the Containment Leakage Rate Testing Program required by 10 CFR 50, Appendix J, and the inservice testing of pumps and valves in accordance with applicable American Society of Mechanical Engineers Operation and Maintenance Code, as required by 10 CFR 50.55a. These programs establish testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations directly or by reference.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been performed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified

BASES

SR 3.0.3 (continued)

Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been completed within the testing interval (including the allowance of SR 3.0.2 if invoked by the Section 5.5 specification).

This delay period provides adequate time to perform Surveillances that have been missed. This delay period permits the performance of a Surveillance before complying with Required Actions or other remedial measures that might preclude performance of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

SR 3.0.3 is only applicable if there is a reasonable expectation the associated equipment is OPERABLE or that variables are within limits, and it is expected that the Surveillance will be met when performed. Many factors should be considered, such as the period of time since the Surveillance was last performed, or whether the Surveillance, or a portion thereof, has ever been performed, and any other indications, tests, or

BASES

SR 3.0.3 (continued)

activities that might support the expectation that the Surveillance will be met when performed. An example of the use of SR 3.0.3 would be a relay contact that was not tested as required in accordance with a particular SR, but previous successful performances of the SR included the relay contact; the adjacent, physically connected relay contacts were tested during the SR performance; the subject relay contact has been tested by another SR; or historical operation of the subject relay contact has been successful. It is not sufficient to infer the behavior of the associated equipment from the performance of similar equipment. The rigor of determining whether there is a reasonable expectation a Surveillance will be met when performed should increase based on the length of time since the last performance of the Surveillance. If the Surveillance has been performed recently, a review of the Surveillance history and equipment performance may be sufficient to support a reasonable expectation that the Surveillance will be met when performed. For Surveillances that have not been performed for a long period or that have never been performed, a rigorous evaluation based on objective evidence should provide a high degree of confidence that the equipment is OPERABLE. The evaluation should be documented in sufficient detail to allow a knowledgeable individual to understand the basis for the determination.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used repeatedly to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important

BASES

SR 3.0.3 (continued)

components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing

BASES

SR 3.0.4 (continued)

to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.9 establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered, unless otherwise specified. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation. Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

BASES

LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The ACTIONS for not meeting a single LCO adequately manage any increase in plant risk, provided any unusual external conditions (e.g., severe weather, offsite power instability) are considered. In addition, the increased risk associated with simultaneous removal of multiple structures, systems, trains or components from service is assessed and managed in accordance with 10 CFR 50.65(a)(4). Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and either:

a. An associated Required Action and Completion Time is not met and no other Condition applies or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

BASES

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. Planned entry into LCO 3.0.3 should be avoided. If it is not practicable to avoid planned entry into LCO 3.0.3, plant risk should be assessed and managed in accordance with 10 CFR 50.65(a)(4), and the planned entry into LCO 3.0.3 should have less effect on plant safety than other practicable alternatives.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to enter lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met,

b. The LCO is no longer applicable,

c. A Condition exists for which the Required Actions have now been performed, or

d. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for entering the next lower MODE applies. If a lower MODE is entered in less time than allowed, however, the total allowable time to enter

BASES

LCO 3.0.3 (continued)

MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is entered in 2 hours, then the time allowed for entering MODE 4 is the next 11 hours, because the total time for entering MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to enter a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.16, "Fuel Storage Pool Water Level." LCO 3.7.16 has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.16 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.16 of "Suspend movement of irradiated fuel assemblies in fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

Spent
3
spent
3
3
spent
1

~~[The requirement to be in MODE 4 in 13 hours is plant specific and depends on the ability to cool the pressurizer and degas.]~~ 2

LCO 3.0.4 LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with either LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

BASES

LCO 3.0.4 (continued)

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered following entry into the MODE or other specified condition in the Applicability will permit continued operation within the MODE or other specified condition for an unlimited period of time. Compliance with ACTIONS that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made and the Required Actions followed after entry into the Applicability.

For example, LCO 3.0.4.a may be used when the Required Action to be entered states that an inoperable instrument channel must be placed in the trip condition within the Completion Time. Transition into a MODE or other specified in condition in the Applicability may be made in accordance with LCO 3.0.4 and the channel is subsequently placed in the tripped condition within the Completion Time, which begins when the Applicability is entered. If the instrument channel cannot be placed in the tripped condition and the subsequent default ACTION ("Required Action and associated Completion Time not met") allows the OPERABLE train to be placed in operation, use of LCO 3.0.4.a is acceptable because the subsequent ACTIONS to be entered following entry into the MODE include ACTIONS (place the OPERABLE train in operation) that permit safe plant operation for an unlimited period of time in the MODE or other specified condition to be entered.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk

BASES

LCO 3.0.4 (continued)

assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Regulatory Guide 1.160 endorses the guidance in Section 11 of NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

BASES

LCO 3.0.4 (continued)

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., RCS Specific Activity), and may be applied to other Specifications based on NRC plant specific approval.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

BASES

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate either:

a. The OPERABILITY of the equipment being returned to service or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance. LCO 3.0.5 should not be used in lieu of other practicable alternatives that comply with Required Actions and that do not require changing the MODE or other specified conditions in the Applicability in order to demonstrate equipment is OPERABLE. LCO 3.0.5 is not intended to be used repeatedly.

An example of demonstrating equipment is OPERABLE with the Required Actions not met is opening a manual valve that was closed to comply with Required Actions to isolate a flowpath with excessive Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) leakage in order to perform testing to demonstrate that RCS PIV leakage is now within limit.

Examples of demonstrating equipment OPERABILITY include instances in which it is necessary to take an inoperable channel or trip system out of a tripped condition that was directed by a Required Action, if there is no Required Action Note for this purpose. An example of verifying OPERABILITY of equipment removed from service is taking a tripped channel out of the tripped condition to permit the logic to function and indicate the appropriate response during performance of required testing on the inoperable channel. Examples of demonstrating the OPERABILITY of other equipment are taking an inoperable channel or trip system out of the tripped condition 1) to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system, or 2) to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

BASES

LCO 3.0.5 (continued)

The administrative controls in LCO 3.0.5 apply in all cases to systems or components in Chapter 3 of the Technical Specifications, as long as the testing could not be conducted while complying with the Required Actions. This includes the realignment or repositioning of redundant or alternate equipment or trains previously manipulated to comply with ACTIONS, as well as equipment removed from service or declared inoperable to comply with ACTIONS.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

BASES

LCO 3.0.6 (continued)

Specification 5.5.14, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

The following examples use Figure B 3.0-1 to illustrate loss of safety function conditions that may result when a TS support system is inoperable. In this figure, the fifteen systems that comprise Train A are independent and redundant to the fifteen systems that comprise Train B. To correctly use the figure to illustrate the SFDP provisions for a cross train check, the figure establishes a relationship between support and supported systems as follows: the figure shows System 1 as a support system for System 2 and System 3; System 2 as a support system for System 4 and System 5; and System 4 as a support system for System 8 and System 9. Specifically, a loss of safety function may exist when a support system is inoperable and:

a. A system redundant to system(s) supported by the inoperable support system is also inoperable (EXAMPLE B 3.0.6-1),

b. A system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2), or

c. A system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).

For the following examples, refer to Figure B 3.0-1.

EXAMPLE B 3.0.6-1

If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in Systems 5, 10, and 11.

EXAMPLE B 3.0.6-2

If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11.

BASES

LCO 3.0.6 (continued)

EXAMPLE B 3.0.6-3

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11.

If an evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

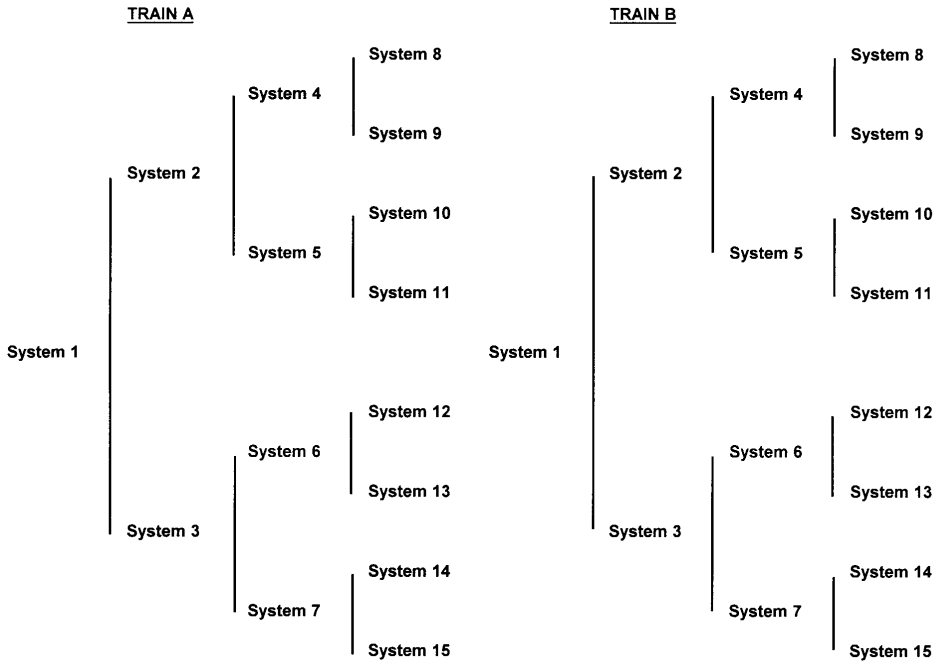

Figure B 3.0-1
Configuration of Trains and Systems

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC

BASES

LCO 3.0.6 (continued)

electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

Special tests and operations are required at various times over the unit's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, special test exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.

A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO

BASES

LCO 3.0.7 (continued)

apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.

Unless the SRs of the specified normal LCOs are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCOs must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

ACTIONS for STE LCOs provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system

BASES

LCO 3.0.8 (continued)

inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

LCO 3.0.9

~~REVIEWER'S NOTE~~

~~Adoption of LCO 3.0.9 requires the licensee to make the following commitments:~~

~~1. [LICENSEE] commits to the guidance of NUMARC 93-01, Revision [4F], Section 11, which provides guidance and details on the assessment and management of risk during maintenance.~~

~~2. [LICENSEE] commits to the guidance of NEI 04-08, "Allowance for Non Technical Specification Barrier Degradation on Supported System OPERABILITY (TSTF-427) Industry Implementation Guidance," March 2006.~~

LCO 3.0.9 establishes conditions under which systems described in the Technical Specifications are considered to remain OPERABLE when required barriers are not capable of providing their related support function(s).

BASES

LCO 3.0.9 (continued)

Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, that support the performance of the safety function of systems described in the Technical Specifications. This LCO states that the supported system is not considered to be inoperable solely due to required barriers not capable of performing their related support function(s) under the described conditions. LCO 3.0.9 allows 30 days before declaring the supported system(s) inoperable and the LCO(s) associated with the supported system(s) not met. A maximum time is placed on each use of this allowance to ensure that as required barriers are found or are otherwise made unavailable, they are restored. However, the allowable duration may be less than the specified maximum time based on the risk assessment.

If the allowed time expires and the barriers are unable to perform their related support function(s), the supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

This provision does not apply to barriers which support ventilation systems or to fire barriers. The Technical Specifications for ventilation systems provide specific Conditions for inoperable barriers. Fire barriers are addressed by other regulatory requirements and associated plant programs. This provision does not apply to barriers which are not required to support system OPERABILITY (see NRC Regulatory Issue Summary 2001-09, "Control of Hazard Barriers," dated April 2, 2001).

The provisions of LCO 3.0.9 are justified because of the low risk associated with required barriers not being capable of performing their related support function. This provision is based on consideration of the following initiating event categories:

~~REVIEWER'S NOTE~~

~~LCO 3.0.9 may be expanded to other initiating event categories provided plant-specific analysis demonstrates that the frequency of the additional initiating events is bounded by the generic analysis or if plant-specific approval is obtained from the NRC.~~

3

- Loss of coolant accidents;
- High energy line breaks;
- Feedwater line breaks;
- Internal flooding;
- External flooding;
- Turbine missile ejection; and
- Tornado or high wind.

BASES

LCO 3.0.9 (continued)

The risk impact of the barriers which cannot perform their related support function(s) must be addressed pursuant to the risk assessment and management provision of the Maintenance Rule, 10 CFR 50.65 (a)(4), and the associated implementation guidance, Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Regulatory Guide 1.160 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This guidance provides for the consideration of dynamic plant configuration issues, emergent conditions, and other aspects pertinent to plant operation with the barriers unable to perform their related support function(s). These considerations may result in risk management and other compensatory actions being required during the period that barriers are unable to perform their related support function(s).

LCO 3.0.9 may be applied to one or more trains or subsystems of a system supported by barriers that cannot provide their related support function(s), provided that risk is assessed and managed (including consideration of the effects on Large Early Release and from external events). If applied concurrently to more than one train or subsystem of a multiple train or subsystem supported system, the barriers supporting each of these trains or subsystems must provide their related support function(s) for different categories of initiating events. For example, LCO 3.0.9 may be applied for up to 30 days for more than one train of a multiple train supported system if the affected barrier for one train protects against internal flooding and the affected barrier for the other train protects against tornado missiles. In this example, the affected barrier may be the same physical barrier but serve different protection functions for each train.

If during the time that LCO 3.0.9 is being used, the required OPERABLE train or subsystem becomes inoperable, it must be restored to OPERABLE status within 24 hours. Otherwise, the train(s) or subsystem(s) supported by barriers that cannot perform their related support function(s) must be declared inoperable and the associated LCOs declared not met. This 24 hour period provides time to respond to emergent conditions that would otherwise likely lead to entry into LCO 3.0.3 and a rapid plant shutdown, which is not justified given the low probability of an initiating event which would require the barrier(s) not capable of performing their related support function(s). During this 24 hour period, the plant risk associated with the existing conditions is assessed and managed in accordance with 10 CFR 50.65(a)(4).

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. SR 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 Specification.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when either:

a. The systems or components are known to be inoperable, although still meeting the SRs or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a special test exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

BASES

SR 3.0.1 (continued)

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

b. High pressure safety injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

BASES

SR 3.0.2 (continued)

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.2 are applicable, a 25% extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted.

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs.

The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Examples of where SR 3.0.2 does not apply are the Containment Leakage Rate Testing Program required by 10 CFR 50, Appendix J, and the inservice testing of pumps and valves in accordance with applicable American Society of Mechanical Engineers Operation and Maintenance Code, as required by 10 CFR 50.55a. These programs establish testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations directly or by reference.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been performed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified

BASES

SR 3.0.3 (continued)

Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

When a Section 5.5, "Programs and Manuals," specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been completed within the testing interval (including the allowance of SR 3.0.2 if invoked by the Section 5.5 specification).

This delay period provides adequate time to perform Surveillances that have been missed. This delay period permits the performance of a Surveillance before complying with Required Actions or other remedial measures that might preclude performance of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

SR 3.0.3 is only applicable if there is a reasonable expectation the associated equipment is OPERABLE or that variables are within limits, and it is expected that the Surveillance will be met when performed. Many factors should be considered, such as the period of time since the Surveillance was last performed, or whether the Surveillance, or a portion thereof, has ever been performed, and any other indications, tests, or

BASES

SR 3.0.3 (continued)

activities that might support the expectation that the Surveillance will be met when performed. An example of the use of SR 3.0.3 would be a relay contact that was not tested as required in accordance with a particular SR, but previous successful performances of the SR included the relay contact; the adjacent, physically connected relay contacts were tested during the SR performance; the subject relay contact has been tested by another SR; or historical operation of the subject relay contact has been successful. It is not sufficient to infer the behavior of the associated equipment from the performance of similar equipment. The rigor of determining whether there is a reasonable expectation a Surveillance will be met when performed should increase based on the length of time since the last performance of the Surveillance. If the Surveillance has been performed recently, a review of the Surveillance history and equipment performance may be sufficient to support a reasonable expectation that the Surveillance will be met when performed. For Surveillances that have not been performed for a long period or that have never been performed, a rigorous evaluation based on objective evidence should provide a high degree of confidence that the equipment is OPERABLE. The evaluation should be documented in sufficient detail to allow a knowledgeable individual to understand the basis for the determination.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used repeatedly to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important

BASES

SR 3.0.3 (continued)

components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing

BASES

SR 3.0.4 (continued)

to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

JUSTIFICATION FOR DEVIATIONS
ITS 3.0, LCO AND SR APPLICABILITY BASES

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is changed to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L01

Florida Power & Light Company (FPL) is converting St. Lucie Plant (PSL) Unit 1 and Unit 2 to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5, "Standard Technical Specifications, Combustion Engineering Plants." The proposed change involves making the Current Technical Specifications (CTS) Less Restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1432.

CTS Section 3.0 does not contain an allowance when barriers cannot support their support function. The proposed change to CTS 3.0, "LCO Applicability" adds a new LCO 3.0.9. The addition of LCO 3.0.9 to the CTS is to address barriers which cannot perform their related support function for Technical Specification systems. ITS LCO 3.0.9 allows barriers to be able to not perform their safety function for up to 30 days before declaring the supported system inoperable. Furthermore, due to this addition, an allowance is also needed in LCO 3.0.1. This allowance has been added.

Barriers are defined as doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications, which are designed to provide for the performance of the safety function for the Technical Specification system after the occurrence of one or more initiating events.

The barrier which cannot perform its related support function will be evaluated and managed under the Maintenance Rule plant configuration control requirement, 10 CFR 50.65(a)(4), and the associated industry guidance (NUMARC 93-01, Revision 4A). This provision is applicable whether the barrier is affected due to planned maintenance or due to a discovered condition. Should the risk assessment and risk management actions for a specific plant configuration or emergent condition not support the 30 day allowed time, the Maintenance Rule risk management determined allowed time and actions must be implemented or the supported system’s LCO be considered not met.

Application of LCO 3.0.9 is dependent on the OPERABILITY of at least one train or subsystem of the supported Technical Specification system and the system's ability to mitigate the consequences of the specified initiating events. However, during the 30 day period allowed by LCO 3.0.9, there exists the possibility that the train or subsystem required to be OPERABLE will unexpectedly become inoperable. Absent any further consideration, this would likely result in both trains of a Technical Specification required system being declared inoperable (i.e., the train supported by the barriers to which LCO 3.0.9 was being applied and the emergent condition of the inoperable train). This would likely result in entering LCO 3.0.3 and a plant shutdown. While this scenario is of low likelihood, it is of very high consequence to the licensee and, therefore, should be avoided unless necessary to avoid an actual plant risk. As a result, LCO 3.0.9 contains a provision which addresses the emergent condition of the required OPERABLE train or subsystem becoming inoperable while LCO 3.0.9 is being used. LCO 3.0.9 provides 24 hours to either restore the inoperable train or subsystem or to cease relying on the provisions of LCO 3.0.9 to consider the train or subsystem supported by the affected barrier(s) OPERABLE. This 24 hour period is not based on a generic risk evaluation, as it would be difficult to perform such an analysis in a generic fashion. Rather, plant risk

during this 24 hour allowance is managed using the contemporaneous risk assessment and management required by 10 CFR 50.65(a)(4) and recognizes the unquantified advantage to plant safety of avoiding a plant shutdown with the associated transition risk.

A risk impact of the 30 day allowance for barriers was performed. PSL initiating events are located on the table depicted in Technical Specifications Task Force (TSTF) traveler TSTF-427, Revision 2, “Allowance for Non Technical Specification Barrier Degradation on Supported System OPERABILITY,“ dated May 3, 2006 (ADAMS Accession No. ML061240055) or PSL has evaluated the use of LCO 3.0.9 for a barrier protecting against an initiating event not on the table located in TSTF-427 and calculated the frequency ranges within the ranges in the table so the above analysis is applicable for those initiators. Therefore, LCO 3.0.9 can be utilized when inoperable barriers affect Systems, Structures, or Components (SSCs).

FPL has evaluated whether a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Barriers are not an initiator to any accident previously evaluated. The probability of an accident previously evaluated is not significantly increased. Barriers support the operation of equipment assumed to mitigate the effects of accidents previously evaluated. The proposed relaxation may only be applied to a single train or subsystem of a multiple train or subsystem Technical Specification system at a given time for a given category of initiating event, or to multiple trains or subsystems of a multiple train or subsystem Technical Specification system provided the affected barriers protect against different categories of initiating events. Therefore, for any given category of initiating event, the ability to perform the assumed safety function is preserved. The consequences of an accident occurring during the time allowed when barriers are not capable of performing their related support function are no different from the consequences of the same accident while relying on the Actions of the supported Technical Specification systems.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   No new or different accidents result from using the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made

in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change allows for a limited period of time in which barriers may be unable to perform their related support function without declaring the supported systems inoperable. A risk analysis has shown that this provision will not have a significant effect on plant risk. In addition, regulatory requirements in 10 CFR 50.65(a)(4) require risk assessment and risk management, which will ensure that plant risk is not significantly increased.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L02

Florida Power & Light Company (FPL) is converting St. Lucie Plant (PSL) Unit 1 and Unit 2 to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5, "Standard Technical Specifications, Combustion Engineering Plants." The proposed change involves making the Current Technical Specifications (CTS) Less Restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1432.

CTS 4.0.2 states, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states, " The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This changes the CTS by adding, " If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance."

This change is acceptable because the 25% Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to Required Actions that must be performed periodically. The initial performance is excluded because the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Completion Time with extension in order to avoid operation in unacceptable conditions. This change is designated as less restrictive because addition time is provided to perform some periodic Required Actions.

FPL has evaluated whether a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change allows the Completion Time for periodic actions to be extended by 25%. This change does not significantly affect the probability of an accident. The length of time between performance of Required Actions is not an initiator to any accident previously evaluated. The consequences of any accident previously evaluated are the same during the Completion Time or during any extension of the Completion Time. As a result, the consequences of any accident previously evaluated are not significantly increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change allows the Completion Time for periodic actions to be extended by 25%. This change will not involve physically altering the plant (i.e., no new or different type of equipment will be installed). In addition, the change does not involve any new or revised operator actions.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change allows the Completion Time for periodic actions to be extended by 25%. The 25% extension allowance is provided for scheduling convenience and is not expected to have significant effect on the average time between Required Actions. As a result, the Required Action will continue to provide appropriate compensatory measures for the subject Condition.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L03

Florida Power & Light Company (FPL) is converting St. Lucie Plant (PSL) Unit 1 and Unit 2 to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, Rev. 5, "Standard Technical Specifications, Combustion Engineering Plants." The proposed change involves making the Current Technical Specifications (CTS) Less Restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1432.

The proposed change revises Sections 3.0, "LCO Applicability" and "SR Applicability" of the Technical Specifications (TS) to clarify the use and application of the TS usage rules and revise the application of Surveillance Requirement (SR) 3.0.3. CTS TS 4.0.3 is revised to allow application of ITS SR 3.0.3 when an SR has not been previously performed.

FPL has evaluated whether a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change to SR 3.0.3 states that the allowance may only be used when there is a reasonable expectation the surveillance will be met when performed. Since the proposed changes does not significantly affect system Operability, the proposed change will have no significant effect on the initiating events for accidents previously evaluated and will have no significant effect on the ability of the systems to mitigate accidents previously evaluated.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change to the TS usage rules does not affect the design or function of any plant systems. The proposed change does not change the Operability requirements for plant systems or the actions taken when plant systems are not operable.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change revised CTS TS 4.0.3 to allow application of ITS SR 3.0.3 when an SR has not been previously performed if there is reasonable expectation that the SR will be met when performed. This expands the use of SR 3.0.3 while ensuring the affected system is capable of performing its safety function. As a result, plant safety is either improved or unaffected..

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified

ENCLOSURE 2

VOLUME 6

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.1
REACTIVITY CONTROL SYSTEMS

Revision 1

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ATTACHMENT 1

3.1.1, SHUTDOWN MARGIN (SDM)

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ITS

A01

ITS 3.1.1

3/~~4.~~1 REACTIVITY CONTROL SYSTEMS

~~3/4.1.1 BORATION CONTROL~~

SHUTDOWN MARGIN ~~- T_{avg} > 200 °F~~

A02

LIMITING CONDITION FOR OPERATION

LCO 3.1.1 3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

See ITS 3.1.6

Applicability **APPLICABILITY**: MODES 1, 2*, 3 and 4.

ACTION:

within 15 minutes

L01

ACTION A With the SHUTDOWN MARGIN not within limits ~~immediately~~ initiate ~~and continue~~ boration ~~at ≥ 40 gpm of greater than or equal to 1900 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored.

L02

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

See ITS 3.1.4

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. When in MODES 1 or 2#, in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

See ITS 3.1.6

c. When in MODE 2## at least once during CEA withdrawal and in accordance with the Surveillance Frequency Control Program until the reactor is critical.

~~d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.~~

L03

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* See Special Test Exception 3.10.1.

\# With $K_{eff} \geq 1.0$.

\## With $K_{eff} < 1.0$.

See ITS 3.1.6

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.1.1.1 Frequency

e. ~~When in MODES 3 or 4,~~ in accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,*
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

LA01

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1000 pcm in accordance with the Surveillance Frequency Control Program. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

See ITS 3.1.2

* For Modes 3 and 4, during calculation of shutdown margin with all CEA's verified fully inserted, the single CEA with the highest reactivity worth need not be assumed to be stuck in the fully withdrawn position.

See ITS 1.1

ITS

ITS 3.1.1

A01

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN ~~- T_{avg} < 200 °F~~

A02

LIMITING CONDITION FOR OPERATION

LCO 3.1.1

3.1.1.2 The SHUTDOWN MARGIN shall be:

Within the limits specified in the COLR, and in addition with the Reactor Coolant System drained below the hot leg centerline, one charging pump shall be rendered inoperable.*

See ITS 3.4.12

Applicability

APPLICABILITY: MODE 5.

ACTION:

within 15 minutes

L01

ACTION A

If the SHUTDOWN MARGIN requirements cannot be met, ~~immediately~~ initiate ~~and continue~~ boration ~~at ≥ 40 gpm of greater than or equal to 1900 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored.

L02

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

4.1.1.2 The SHUTDOWN MARGIN requirements ~~of Specification 3.1.1.2~~ shall be determined:

See ITS 3.1.4

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

LA01

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2.b and by verifying at least one charging pump is rendered inoperable.*

See ITS 3.4.12

* Breaker racked-out.

See ITS 3.4.12

ITS

A01

ITS 3.1.1

3/4.1 REACTIVITY CONTROL SYSTEMS

~~3/4.1.1 BORATION CONTROL~~

SHUTDOWN MARGIN ~~- T_{avg} GREATER THAN 200°F~~

A02

LIMITING CONDITION FOR OPERATION

LCO 3.1.1

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

See ITS 3.1.6

Applicability

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

within 15 minutes

L01

ACTION A

With the SHUTDOWN MARGIN outside the COLR limits, ~~immediately~~ initiate ~~and continue~~ boration ~~at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored.

L02

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

See ITS 3.1.4

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

See ITS 3.1.6

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

See ITS 3.1.6

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

~~d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.~~

L03

SR 3.1.1.1 Frequency

e. ~~When in MODE 3 or 4~~, in accordance with the Surveillance Frequency Control Program by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

LA01

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1000 pcm in accordance with the Surveillance Frequency Control Program. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPDs after each fuel loading.

See ITS 3.1.2

ITS

ITS 3.1.1

REACTIVITY CONTROL SYSTEMS

A01

SHUTDOWN MARGIN - ~~T_{avg} LESS THAN OR EQUAL TO 200°F~~

A02

LIMITING CONDITION FOR OPERATION

LCO 3.1.1 3.1.1.2 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

Applicability **APPLICABILITY**: MODE 5.

ACTION:

within 15 minutes

L01

ACTION A With the SHUTDOWN MARGIN outside the COLR limits, ~~immediately~~ initiate ~~and continue~~ boration ~~at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored.

L02

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 4.1.1.2 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

See ITS 3.1.4

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

LA01

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2b and by verifying at least two charging pumps are rendered inoperable by racking out their motor circuit breakers.

See ITS 3.4.12

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.1.1.1 provides the SHUTDOWN MARGIN (SDM) requirement in MODES 1, 2, 3, and 4 (i.e., $T_{avg} > 200°F$). CTS 3.1.1.2 provides the SDM requirement in MODE 5 (i.e., $T_{avg} \leq 200°F$). ITS 3.1.1 provides the SDM requirement in MODES 3, 4, and 5. This changes the CTS by combining the SDM requirements in MODES 3, 4, and 5. The Applicability for MODES 1 and 2 are described in ITS 3.1.6, “Regulating Control Element Assembly (CEA) Insertion Limits.”

This change is acceptable because the requirements have not changed. Combining the Specifications is an editorial change. Any technical changes resulting from this combination are discussed in other DOCs. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.1.1.1.1.e and CTS 4.1.1.2.b require determination that the SDM is within limits, and specifically requires the consideration of the following factors: reactor coolant system boron concentration, CEA position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration and samarium concentration. ITS SR 3.1.1.1 requires a determination that the SDM is within limits, but does not describe the factors that must be considered in the calculation. This information is moved to the Bases. This changes the CTS by removing details on how the SDM calculation is performed from the Specification and placing the information in the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement that the SDM be within limits. The detail of how SDM is calculated does not need to appear in the specification in order for the requirement to apply. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.1.1.1 ACTION states when the SDM is less than the applicable limit, boration must be initiated immediately. ITS 3.1.1 ACTION states when SDM is not within limits, boration must be initiated within 15 minutes. This changes the CTS by relaxing the Completion Time from "immediately" to 15 minutes.

The purpose of CTS 3.1.1.1 ACTION is to restore the SDM to within its limit promptly. This change is acceptable because the Completion Time is consistent with safe operation under the specific Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, and the low probability of a DBA occurring during the allowed Completion Time. This ITS Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. In addition, the ITS Bases for the ACTION states that boration must be initiated promptly. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.1.1.1 ACTION states when the SDM is not within limits, boration must be initiated and continued at ≥ 40 gpm of a solution containing greater than or equal to 1900 ppm boron or equivalent until the required SDM is restored. ITS 3.1.1 ACTION A states that when the SDM is not within limits to initiate boration to restore SDM to within limits. This changes the CTS by eliminating the specific values of flow rate and the boron concentration used to restore compliance with the LCO.

The purpose of CTS 3.1.1.1 ACTION is to restore the SDM to within its limit. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the specified redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required

features, and the low probability of a DBA occurring during the allowed Completion Time. Removing the specific values of flow rate and boron concentration from the CTS ACTION provides flexibility in the restoration of the SDM and eliminates conflicts between the SDM value and the specific boration values in the CTS ACTION. As stated, in the ITS Bases for ACTION A, "In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid makeup tank, or the refueling water tank. The operator should borate with the best source available for the plant conditions." Specifying a minimum flow rate and concentration in the ACTION may not accomplish the objective of raising the RCS boron concentration as soon as possible. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.1.1.1.d requires verification that the SDM is within limit, "Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below [CTS 4.1.1.1.1.e], with the CEA groups at the power dependent insertion limits of Specification 3.1.3.6." The ITS does not contain a similar requirement. This changes the CTS by deleting Surveillance Requirement 4.1.1.1.1.d.

The purpose of CTS 4.1.1.1.1.d is to verify core design predictions by determining the SDM with the CEAs at the insertion limits. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify the LCO is within limit. The core design predictions, such as rod worth, boron worth, and critical boron concentration, are verified in a manner and at a Frequency necessary to give confidence that these predicted values are within limit in accordance with ITS SR 3.1.2.1. ITS SR 3.1.2.1 has a conditional Frequency similar to that of CTS 4.1.1.1.d requiring performance prior to entering MODE 1 (> 5% RTP) after fuel loading. To ensure the SDM is within limits during reactor startup the critical boron concentration is verified during the startup physics test program. Thereafter SDM is confirmed by performance of ITS SR 3.1.4.1 (CEA alignment verification), SR 3.1.5.1(shutdown CEA withdrawn verification), and SR 3.1.6.1 (regulating CEA group position verification). Thus, the SDM continues to be verified in a manner and at a Frequency necessary to give confidence that the parameter is within limit. Therefore, the core design parameters upon which SDM relies are verified before exceeding 5% RATED THERMAL POWER after each fuel loading. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~ (1)

3.1.1 SHUTDOWN MARGIN (SDM) ~~(Analog)~~

3.1.1.1
3.1.1.2 LCO 3.1.1 SDM shall be within the limits specified in the COLR.

Applicability APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits.	15 minutes

Action
DOC L01
DOC L02

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM to be within the limits specified in the COLR.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)

4.1.1.1.e
4.1.1.2.b

CTS

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

3.1.1 SHUTDOWN MARGIN (SDM) ~~(Analog)~~ (1)

3.1.1.1
3.1.1.2

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

Applicability

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits.	15 minutes

Action
DOC L01
DOC L02

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM to be within the limits specified in the COLR.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)

4.1.1.1.e
4.1.1.2.b

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.1, SHUTDOWN MARGIN

1. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~ 1

B 3.1.1 SHUTDOWN MARGIN (SDM) ~~(Analog)~~

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SHUTDOWN MARGIN (SDM) ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the CEA of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth CEA stuck out following a reactor trip. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events,

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limit AOOs, and ≤ 2~~8~~0 cal/gm energy deposition for the CEA ejection accident), and 3 2

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements are based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3 and 4 must also protect against:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. Inadvertent boron dilution,

b. An uncontrolled CEA withdrawal from a subcritical condition,

~~c. Startup of an inactive reactor coolant pump (RCP),~~ and

c

~~d~~. CEA ejection. 2

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

The withdrawal of CEAs from subcritical conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a high power trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

~~The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. An idle RCP cannot, therefore, produce a return to power from the hot standby condition.~~

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

2

The ejection of a CEA rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a CEA also produces a time dependent redistribution of core power.

BASES

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR ~~100, "Reactor Site Criteria~~," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

50.67, "Accident source term 2

SDM is a core physics design condition that can be ensured through CEA positioning (regulating and shutdown CEA) and through the soluble boron concentration.

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.6. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid makeup ~~storage~~ tank or the ~~borated~~ water ~~storage~~ tank. The operator should borate with the best source available for the plant conditions.

refueling

2

in

4

In determining the boration flow rate, the time core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. ~~Assuming that a value of 1% Δk/k must be recovered and a~~ A boration flow rate of ~~[ ]~~ 40 gpm and ~~[ ]~~ a boron concentration of 1900 ppm represent typical values ~~and are provided for the purpose of offering a specific example~~. needed to recover SDM to within limits

5

3

3

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration,

b. CEA positions,

c. RCS average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration,

f. Samarium concentration~~,~~ and

~~g. Isothermal temperature coefficient (ITC)~~.

~~Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.~~ 2

~~[The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 6

~~]~~

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section ~~[ ]~~. 15.4.6
3. FSAR, Section ~~[ ]~~. 15.2.4
4. 10 CFR ~~100~~. 50.67

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B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~ 1

B 3.1.1 SHUTDOWN MARGIN (SDM) ~~(Analog)~~

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SHUTDOWN MARGIN (SDM) ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the CEA of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth CEA stuck out following a reactor trip. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events,

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limit AOOs, and ≤ 2~~8~~80 cal/gm energy deposition for the CEA ejection accident), and 3 2

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements are based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3 and 4 must also protect against:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. Inadvertent boron dilution,

b. An uncontrolled CEA withdrawal from a subcritical condition,

~~c. Startup of an inactive reactor coolant pump (RCP),~~ and

c

~~d~~. CEA ejection. 2

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

The withdrawal of CEAs from subcritical conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a high power trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

~~The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. An idle RCP cannot, therefore, produce a return to power from the hot standby condition.~~

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

2

The ejection of a CEA rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a CEA also produces a time dependent redistribution of core power.

BASES

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR ~~100, "Reactor Site Criteria~~," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

50.67, "Accident source term 2

SDM is a core physics design condition that can be ensured through CEA positioning (regulating and shutdown CEA) and through the soluble boron concentration.

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.6. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid makeup ~~storage~~ tank or the ~~borated~~ water ~~storage~~ tank. The operator should borate with the best source available for the plant conditions.

refueling

2

in

4

In determining the boration flow rate, the time core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. ~~Assuming that a value of 1% Δk/k must be recovered and a~~ boration flow rate of ~~[ ] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of [ ]~~ gpm and ~~[ ]~~ ppm represent typical values ~~and are provided for the purpose of offering a specific example~~.

A

5

3

needed to recover SDM to within limits

40

a boron concentration of 1900

3

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration,

b. CEA positions,

c. RCS average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration,

f. Samarium concentration~~,~~ and

~~g. Isothermal temperature coefficient (ITC)~~.

~~Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.~~ 2

~~[The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 6

~~]~~

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section ~~[ ]~~. 15.4.6
3. FSAR, Section ~~[ ]~~. 15.2.4
4. 10 CFR ~~100~~. 50.67

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JUSTIFICATION FOR DEVIATIONS
ITS 3.1.1 BASES, SHUTDOWN MARGIN (SDM)

1. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. Editorial changes made for enhanced clarity/consistency or grammatical error corrected.

5. As stated, the ISTS Bases discussion regarding the required boration flow rate and time necessary to recover 1% $\Delta k/k$ of SDM is provided for the purpose of offering a specific example. St. Lucie Plant operating procedures provide guidance to determine the necessary flow rate and boron concentration to recover SDM based on plant conditions. Therefore, the example provided in the ISTS 3.1.1 Bases is not included in the plant-specific ITS 3.1.1 Bases.

6. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

3.1.2, Reactivity Balance

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ITS

A01

ITS 3.1.2

3/~~4.~~1 REACTIVITY CONTROL SYSTEMS

~~3/4.1.1 BORATION CONTROL~~

Reactivity Balance

A02

~~SHUTDOWN MARGIN - T_{avg} > 200 °F~~

Add proposed LCO 3.1.2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

See ITS 3.1.1

Applicability **APPLICABILITY**: MODES 1, 2*, ~~3 and 4.~~

See ITS 3.1.6

L01

ACTION:

Add proposed ACTIONS A and B

L02

With the SHUTDOWN MARGIN not within limits immediately initiate and continue boration at ≥ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

See ITS 3.1.1

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

See ITS 3.1.4

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. When in MODES 1 or 2#, in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

See ITS 3.1.6

c. When in MODE 2## at least once during CEA withdrawal and in accordance with the Surveillance Frequency Control Program until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

See ITS 3.1.1

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* See Special Test Exception 3.10.1.

\# With K_{eff} ≥ 1.0.

With K_{eff} < 1.0.

See ITS 3.1.6

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, in accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,*
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

Add proposed Note 2

See ITS 3.1.1

Prior to entering MODE 1 after fuel loading AND

L03

1% Δk/k

A02

Note Only required after 60 EFPD

LA01

LCO 3.1.2
SR 3.1.2.1

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± ~~1000 pcm~~ in accordance with the Surveillance Frequency Control Program. ~~This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above.~~ The predicted reactivity values ~~shall~~ be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

SR 3.1.2.1
Note 1

may

L04

* For Modes 3 and 4, during calculation of shutdown margin with all CEA's verified fully inserted, the single CEA with the highest reactivity worth need not be assumed to be stuck in the fully withdrawn position.

See ITS 1.1

ITS

ITS 3.1.2

A01

3/4.1 REACTIVITY CONTROL SYSTEMS

~~3/4.1.1 BORATION CONTROL~~

Reactivity Balance

A02

~~SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F~~

Add proposed LCO 3.1.2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

See ITS 3.1.1

Applicability

APPLICABILITY: MODES 1, 2*, ~~3 and 4~~.

See ITS 3.1.6

L01

ACTION:

Add proposed ACTIONS A and B

L02

With the SHUTDOWN MARGIN outside the COLR limits, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

See ITS 3.1.1

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

See ITS 3.1.4

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

See ITS 3.1.6

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

See ITS 3.1.6

ITS

A01

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

See ITS 3.1.1

e. When in MODE 3 or 4, in accordance with the Surveillance Frequency Control Program by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

See ITS 3.1.1

Add proposed Note 2

LCO 3.1.2
SR 3.1.2.1

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± ~~1000 pcm~~ in accordance with the Surveillance Frequency Control Program. ~~This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above~~. The predicted reactivity values ~~shall~~ be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPDs after each fuel loading.

1% Δk/k

A02

Prior to entering MODE 1 after fuel loading AND

L03

Note Only required after 60 EFPD

LA01

SR 3.1.2.1
Note 1

may

L04

DISCUSSION OF CHANGES
ITS 3.1.2, REACTIVITY BALANCE

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 4.1.1.1.2 requires the overall core reactivity balance to be compared to predicted values to demonstrate agreement within ± 1000 pcm. However, this Surveillance is currently part of the SHUTDOWN MARGIN Specification. Additionally, CTS 3.1.1.1 is titled SHUTDOWN MARGIN – T_{avg} > 200°F. A new LCO, ITS LCO 3.1.2, requires the measured core reactivity balance to be within ± 1% Δk/k of predicted values. Furthermore, ITS 3.1.2 is titled Reactivity Balance. This changes the CTS by having a separate Specification for the core reactivity requirement and changing the title. This change also changes the units of 1000 pcm to 1% Δk/k which are equivalent units of reactivity.

This change is acceptable because the requirements have not changed. Converting the requirement from a Surveillance in the SHUTDOWN MARGIN specification to an LCO is consistent with the ITS format and content guidance. Any technical changes resulting from this change are discussed in other DOCs. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.1.1.1.2 requires comparison of the actual and predicted core reactivity balance and specifically requires consideration of at least those factors stated in Specification 4.1.1.1.1.e. CTS 4.1.1.1.1.e requires determination of SDM and requires the consideration of the following factors: reactor coolant system boron concentration, control element assembly (CEA) position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration when in MODES 3 or 4. ITS SR 3.1.2.1 requires overall core

reactivity balance to be within the predicted values, but does not describe the factors that must be considered in the calculation. This information is relocated to the Bases. This changes the CTS by removing details on how the core reactivity balance comparison calculation is performed from the CTS and placing the information in the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.1.2 still retains the requirement that the core reactivity balance comparison be within ± 1% $\Delta k/k$ in MODES 1 and 2. Refer to Discussion of Change L02 associated with deletion of MODES 3 and 4 from Applicability of CTS 4.1.1.1.2.

The details of how this comparison is calculated do not need to appear in the Specification in order for the requirement to apply. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 4.1.1.1.2 is applicable in MODES 1, 2, 3, and 4. ITS 3.1.2 is applicable in MODES 1 and 2. This changes the CTS by reducing the applicable MODES in which the core reactivity requirement must be met.

The purpose of CTS Surveillance 4.1.1.1.2 is to verify the core design by comparing the actual and predicted core reactivity. This change is acceptable because the requirements continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analysis and licensing basis. The core reactivity balance can only be determined when the reactor is critical (MODES 1 and 2). Reducing the applicable MODES from MODES 1, 2, 3, and 4 to MODES 1 and 2 does not result in a reduction of the verification of this important measure of core design accuracy. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.1.1.1 does not contain ACTIONS to follow if the core reactivity balance Surveillance is not met. If the core reactivity balance Surveillance is not met, CTS 3.0.3 would be entered. CTS 3.0.3 requires action to be initiated within 1 hour and the unit to be in hot standby (MODE 3) within the next 6 hours, hot shutdown (MODE 4) within the following 6 hours, and cold shutdown (MODE 5) within the subsequent 24 hours. ITS 3.1.2 contains ACTIONS to follow if the core reactivity balance LCO is not met. If the LCO is not met, 7 days are provided to re-evaluate the core design

and safety analysis, to determine that the reactor core is acceptable for continued operation, and to establish appropriate operating restrictions and SRs. If these actions are not completed within the 7 days, the plant must be placed in MODE 3 within 6 hours. This changes the CTS by providing 7 days to evaluate and provide compensatory measures for not meeting the core reactivity balance requirement and then requiring entry into MODE 3 instead of requiring a unit shutdown and entry into MODE 5.

The purpose of CTS 4.1.1.1.2 is to verify the accuracy of the core design by comparing the predicted and actual core reactivity throughout core life. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Should the core reactivity balance requirement not be met, time is required to determine the cause of the disagreement and what adjustments may be needed to the operating conditions of the core. The startup physics testing program is used to verify most of the critical core design parameters, such as CEA worth, boron worth, and moderator temperature coefficient. In addition, there is considerable conservatism in the application of these values in the accident analyses. Therefore, allowing a time to evaluate the difference and make any adjustments to the operational controls is acceptable. The 7 day Completion time is reasonable considering the complexity of the evaluations and the time to meet administrative requirements, such as 10 CFR 50.59 safety evaluation preparation and approval. If it cannot be determined within 7 days that the core is acceptable for continued operation, the unit must be shutdown. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.1.1.1.2 requires comparison of the actual and predicted core reactivity balance in accordance with the Surveillance Frequency Control Program. CTS 4.1.1.1.2 also requires the predicted reactivity values to be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. ITS SR 3.1.2.1 requires verifying the overall core reactivity balance is within ± 1 % Δ k/k of the predicted values prior to entering MODE 1 after each refueling and in accordance with the Surveillance Frequency Control Program after 60 EFPD. Note 2 to ITS SR 3.1.2.1 states that the Surveillance is not required to be performed prior to entry into MODE 2. This changes the CTS by not requiring the periodic, at-power core reactivity comparison until core burnup reaches 60 EFPD. Additionally, it allows the initial verification following fuel loading to be performed after entering MODE 2 and prior to MODE 1.

The purpose of CTS 4.1.1.1.2 is to verify the agreement between the actual and predicted core reactivity. The CTS requires and the ITS may require the predicted core reactivity values to be normalized to the actual values prior to exceeding 60 EFPD of core burnup. This allows sufficient time for core

conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The change allowing the required subsequent periodic Frequency in accordance with the Surveillance Frequency Control Program to be performed after entering MODE 2 and following the initial 60 EFPD after fuel loading, is acceptable, based on the slow rate of core reactivity changes resulting from fuel depletion and the presence of other indicators (AZIMUTHAL POWER TILT, AXIAL SHAPE INDEX, etc.) for prompt indication of an anomaly. This change has been designated as less restrictive because the Surveillance will be performed in a different MODE of operation under the ITS than under the CTS.

L04 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)*
CTS 4.1.1.1.2 requires, in part, that the predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading. ITS SR 3.1.2.1 Note 1 states the adjustment "may" be performed prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. This changes the CTS by stating that the normalization may be performed prior to 60 EFPD after each fuel loading.

The purpose of adjusting the predicted reactivity values to the core conditions is to allow benchmarking of the design calculations. Making this adjustment closer to a fuel burnup of 60 EFPD of operation allows sufficient time for the core conditions to reach steady state. This change is acceptable because the expectation is to perform the adjusting of the predicted reactivity values to the core conditions. ITS SR 3.1.2.1 still allows the adjustment to take place prior to the 60 EFPD after each fuel loading. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

3.1.2 Reactivity Balance ~~(Analog)~~ 1

4.1.1.1.2 LCO 3.1.2 The core reactivity balance shall be within ± 1% Δk/k of predicted values.

DOC A02

DOC L01 APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L02 A. Core reactivity balance not within limit.	A.1 Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation. AND A.2 Establish appropriate operating restrictions and SRs.	7 days 7 days
DOC L02 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.1.1.1.2 SR 3.1.2.1	----------NOTES---------- 1. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. DOC L04 2. This Surveillance is not required to be performed prior to entry into MODE 2. ---------- Verify overall core reactivity balance is within ± 1.0% Δk/k of predicted values.	Prior to entering MODE 1 after fuel loading DOC L03 AND --------NOTE-------- Only required after 60 EFPD ---------- ~~[31 EFPD~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 3

CTS

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

3.1.2 Reactivity Balance ~~(Analog)~~ 1

4.1.1.1.2 LCO 3.1.2 The core reactivity balance shall be within ± 1% Δk/k of predicted values.

DOC A02

DOC L01 APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L02 A. Core reactivity balance not within limit.	A.1 Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation. AND A.2 Establish appropriate operating restrictions and SRs.	7 days 7 days
DOC L02 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.1.1.1.2 SR 3.1.2.1	----------NOTES---------- 1. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. DOC L04 2. This Surveillance is not required to be performed prior to entry into MODE 2. ---------- Verify overall core reactivity balance is within ± 1.0% Δk/k of predicted values.	Prior to entering MODE 1 after fuel loading DOC L03 AND --------NOTE-------- Only required after 60 EFPD ---------- ~~[31 EFPD~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.2, REACTIVITY BALANCE

1. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

1

B 3.1.2 Reactivity Balance ~~(Analog)~~

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that, subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control element assembly (CEA) worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the critical boron curve, which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as CEA height, temperature, pressure, and power) provides a convenient method of ensuring that core reactivity is within design expectations, and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), CEAs, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

BASES

BACKGROUND (continued)

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The critical boron curve is based on steady state operation at RTP. Therefore, deviations from the predicted critical boron curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as CEA withdrawal accidents or CEA ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted critical boron curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the CEAs in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

The reactivity balance satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The reactivity balance limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta k/k$ has been established, based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% $\Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, or CEA replacement, or shuffling).

BASES

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized, and power operation may continue. If operational restrictions or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable including CEA position, ~~moderator~~ RCS boron concentration, RCS average temperature, ~~fuel temperature,~~ fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by three Notes. Note 1 in the Surveillance column indicates that the normalization of predicted core reactivity to the measured value ~~must~~ take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. ~~[The required subsequent Frequency of 31 EFPD following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g., QPTR, etc.) for prompt indication of an anomaly.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

A second Note, "only required after 60 EFPD," is added to the Frequency column to allow this. Note 2 in the Surveillance column indicates that the performance of SR 3.1.2.1 is not required prior to entering MODE 2. This Note is required to allow a MODE 2 entry to verify core reactivity, because LCO Applicability is for MODES 1 and 2.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
2. FSAR, Section ~~[~~ [4.3~~]~~.

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B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

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B 3.1.2 Reactivity Balance ~~(Analog)~~

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that, subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control element assembly (CEA) worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the critical boron curve, which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as CEA height, temperature, pressure, and power) provides a convenient method of ensuring that core reactivity is within design expectations, and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), CEAs, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

BASES

BACKGROUND (continued)

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The critical boron curve is based on steady state operation at RTP. Therefore, deviations from the predicted critical boron curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as CEA withdrawal accidents or CEA ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the CEAs in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

The reactivity balance satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The reactivity balance limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta k/k$ has been established, based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% $\Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, or CEA replacement, or shuffling).

BASES

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized, and power operation may continue. If operational restrictions or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable including CEA position, ~~moderator~~ RCS boron concentration, RCS average temperature, ~~fuel temperature,~~ fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by three Notes. Note 1 in the Surveillance column indicates that the normalization of predicted core reactivity to the measured value ~~must~~ take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. ~~[The required subsequent Frequency of 31 EFPD following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g., QPTR, etc.) for prompt indication of an anomaly.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

A second Note, "only required after 60 EFPD," is added to the Frequency column to allow this. Note 2 in the Surveillance column indicates that the performance of SR 3.1.2.1 is not required prior to entering MODE 2. This Note is required to allow a MODE 2 entry to verify core reactivity, because LCO Applicability is for MODES 1 and 2.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
2. FSAR, Section ~~[~~ [4.3~~]~~.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.1.2 BASES, REACTIVITY BALANCE

1. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. Changes are made to be consistent with the requirements of the Specification.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.2, REACTIVITY BALANCE

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

3.1.3, Moderator Temperature Coefficient (MTC)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

LCO 3.1.3

3.1.1.4 The moderator temperature coefficient (MTC) shall be maintained within the limits specified in the COLR. The maximum positive limit shall be:

a. ~~Less positive than~~ +7 pcm/°F when~~ever~~ THERMAL POWER is ≤ 70% of RATED THERMAL POWER, and

b. ~~Less positive than~~ +2 pcm/°F when~~ever~~ THERMAL POWER is > 70% of RATED THERMAL POWER.

A02

M01

Applicability

APPLICABILITY: MODES 1 AND 2~~*#~~.

A03

ACTION:

ACTION A

With the moderator temperature coefficient outside any one of the above limits, be in HOT STANDBY within 6 hours.

A02

the COLR

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

4.1.1.4.1 Verify MTC is within the upper limit specified in ~~LCO 3.1.1.4~~.

a. Prior to entering MODE 1 after each fuel loading, and

b. Each fuel cycle within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup.**

M01

~~* With K_{eff} ≥ 1.0~~.

SR 3.1.3.1
2nd Freq. Note

** Only required to be performed when MTC determined prior to entering MODE 1 is verified using adjusted predicted MTC.

~~# See Special Test Exception 3.10.2.~~

A03

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.2 4.1.1.4.2*** Verify MTC is within the lower limit specified in the COLR.**** | R1

Each fuel cycle within 7 EFPD of reaching 2/3 of expected core burnup.

SR 3.1.3.2 Note 2 *** If MTC is more negative than the lower limit specified in the COLR when extrapolated to the end of cycle, 4.1.1.4.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit.

to be performed

SR 3.1.3.2 Note 1 **** Only Required if ~~the~~ MTC determined ~~in SR 4.1.1.4.1~~ is not within ~~±~~ 1.6 pcm/°F of ~~the corresponding design value~~.

predicted MTC

during performance of SR 3.1.3.1

A04

A01

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

LCO 3.1.3 3.1.1.4 The moderator temperature coefficient (MTC) shall be maintained within the limits specified in the COLR. The maximum ~~upper~~ limit shall be +5 pcm/°F at ≤ 70% of RATED THERMAL POWER, with a linear ~~ramp~~ from +5 pcm/°F at 70% of RATED THERMAL POWER to 0 pcm/°F at 100% RATED THERMAL POWER.

positive

decrease

M01

Applicability **APPLICABILITY**: MODES 1 AND 2*#.

A03

ACTION:

ACTION A With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

A02

the COLR

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 4.1.1.4.1 Verify MTC is within the upper limit specified in ~~LCO 3.1.1.4~~.

a. Prior to entering MODE 1 after each fuel loading, and

b. Each fuel cycle within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup. **

SR 3.2.1.3 4.1.1.4.2*** Verify MTC is within the lower limit specified in the COLR.****

Each fuel cycle within 7 EFPD of reaching 2/3 of expected core burnup.

A03

~~# See Special Test Exception 3.10.2 and 3.10.5.~~

M01

~~* With K_{eff} greater than or equal to 1.0.~~

SR 3.1.3.1 2nd Freq. Note ** Only required to be performed when MTC determined prior to entering MODE 1 is verified using adjusted predicted MTC.

SR 3.1.3.2 Note 2 *** If MTC is more negative than the lower limit specified in the COLR when extrapolated to the end of cycle, 4.1.1.4.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit.

to be performed

SR 3.1.3.2 Note 1 **** Only Required if ~~the~~ MTC determined ~~in SR 4.1.1.4.1~~ is not within ~~±~~ 1.6 pcm/°F of ~~the corresponding design value~~.

predicted MTC

during performance of SR 3.1.3.1

A04

DISCUSSION OF CHANGES
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.1.1.4 states, in part, that the maximum positive limit shall be less positive than +7 pcm/°F whenever THERMAL POWER is ≤ 70% of RATED THERMAL POWER, and less positive than +2 pcm/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER. Unit 1 ITS 3.1.3 states, in part, that the maximum positive limit shall be +7 pcm/°F when THERMAL POWER is ≤ 70% RTP, and +2 pcm/°F when THERMAL POWER is > 70% RTP. This changes the Unit 1 CTS by eliminating the wording “less positive than,” prior to listing the maximum positive MTC limit. This change is acceptable because the Unit 1 LCO requires a maximum value not a variable maximum value. The first sentence of the LCO requires the MTC to be maintained within the limits specified in the COLR and the second sentence is intended to ensure the upper MTC value specified in the COLR does not exceed a maximum positive MTC as stated in the LCO. In addition, Unit 1 and Unit 2 CTS 4.1.1.4.1, which requires verification that MTC is within the upper MTC limit, is changed to reference the COLR instead of the LCO, which references the maximum positive (i.e., upper) MTC limit. The limiting upper MTC specified in the COLR may be more restrictive than the maximum positive limit specified in the LCO. Therefore, ITS SR 3.1.3.1 requires verifying the MTC is within the upper limit specified in the COLR, which would encompass verifying the MTC is within the positive upper limit specified in the LCO. These changes are designated as administrative because they represent a presentation preference and do not represent a technical change to the CTS.

A03 The Applicability of CTS 3.1.1.4 is modified by footnote # stating "See Special Test Exception 3.10.2 (and 3.10.5 for Unit 2 only)." ITS 3.1.3 Applicability does not contain the footnote or a reference to the Special Test Exception. This changes the CTS by not including footnote # in the ITS.

The purpose of the footnote reference is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

A04 CTS 4.1.1.4.2 requires verification that the lower MTC is within the limits of the COLR. ****Note states that this Surveillance is only required if the MTC determined in SR 4.1.1.4.1 is not within ± 1.6 pcm/°F of the corresponding design value. ITS SR 3.1.3.2 contains a similar note (Note 1), which states, “Only required to be performed if MTC determined during performance of SR 3.2.3.1 is

not within 1.6 pcm/°F of the predicted MTC value." This changes the CTS by revising the Note to clarify the intent that the 1.6 pcm/°F criterion is based on the difference between the MTC determined during the performance of CTS 4.1.1.4.1 (ITS SR 3.1.3.1) and the predicted MTC at the point in core life when the Surveillance is performed.

In September 2016, the CTS **** Note was added in Amendments 235 and 185 for PSL Units 1 and 2, respectively, to waive performance of the MTC end of cycle measurement at or near 2/3 of expected core burn up if the MTC determined during performance of CTS 4.1.1.4.1 is within 1.6 pcm/°F of the predicted value (NRC ADAMS Accession No. ML16183A138). This change in presentation is designated as administrative because it only provides clarification to the original intent as approved in Amendments 235 (Unit 1) and 185 (Unit 2) and does not represent a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.1.1.4 is applicable in MODES 1 and 2 with an * Note that states, "With K_{eff} ≥ 1.0 (Unit 1 CTS)," and "With K_{eff} greater than or equal to 1.0 (Unit 2 CTS." ITS 3.1.3 is applicable in MODES 1 and 2. This changes the CTS by requiring MTC to be maintained within limits in MODE 2 when k_{eff} is < 1.0.

MTC is an input assumption in the PSL accident analyses, particularly core overheating and core overcooling events. MTC is also an input assumption in startup and subcritical events, such as the uncontrolled CEA withdrawal from a subcritical or low power startup condition. Therefore, MTC must be maintained within limits during MODE 2, whether the reactor is critical or subcritical. This change is consistent with the Applicability of the ISTS and is designated as more restrictive because the Applicability of the CTS LCO requirements is expanded to include subcritical conditions while in MODE 2.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

Note

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

3.1.3 Moderator Temperature Coefficient (MTC) ~~(Analog)~~

1

3.1.1.4 LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum positive limit shall be: ~~that specified in Figure 3.1.3-1.~~

DOC A02

a. +7 pcm/°F with THERMAL POWER ≤ 70% RTP, and

b. +2 pcm/°F with THERMAL POWER > 70% RTP.

2

Applicability APPLICABILITY: MODES 1 and 2.

DOC M01
DOC A03

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limits.	A.1 Be in MODE 3.	6 hours

Action

SURVEILLANCE REQUIREMENTS

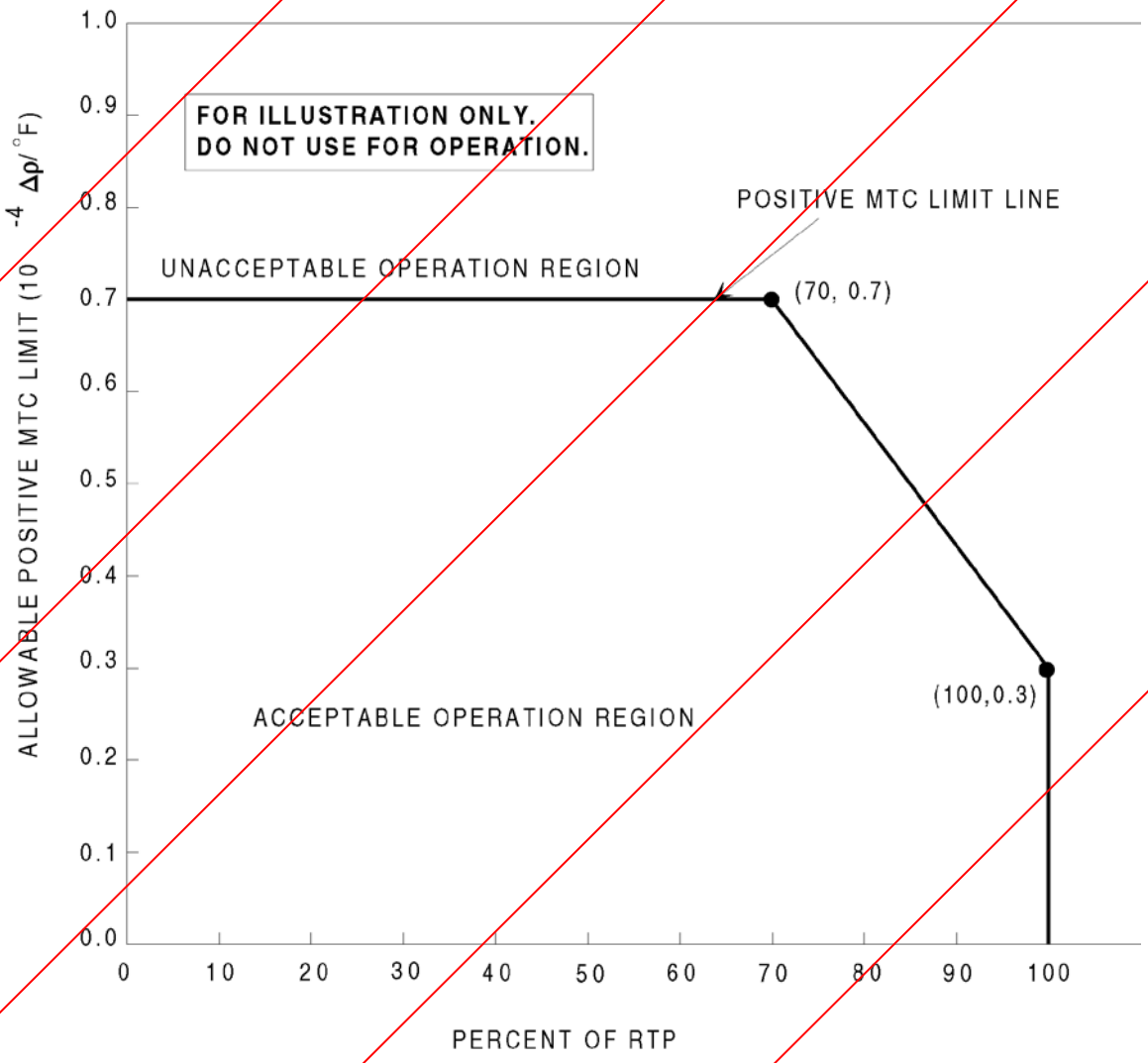
	SURVEILLANCE		FREQUENCY
4.1.1.4.1 4.1.1.4.1.a	SR 3.1.3.1	Verify MTC is within the upper limit specified in the COLR.	Prior to entering MODE 1 after each fuel loading ~~[~~ AND 4 --------NOTE-------- **Note Only required to be performed when MTC determined prior to entering MODE 1 is verified using adjusted predicted MTC ---------- 4.1.1.4.1.b Each fuel cycle within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	----------NOTES----------	5
***Note	If ~~the~~ MTC is more negative than the limit specified in the COLR when extrapolated to the end of cycle, SR 3.1.3.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit.	

4.1.1.4.2	Verify MTC is within the lower limit specified in the COLR.	~~Each fuel cycle within 7 EFPD of reaching 40 EFPD core burnup~~ 6 ~~AND~~
****Note DOC A04	1. Only required to be performed if MTC determined during performance of SR 3.1.3.1 is not within 1.6 pcm/°F of predicted MTC. 2.	Each fuel cycle within 7 EFPD of reaching 2/3 of expected core burnup 5

2



~~Figure 3.1.3-1 (page 1 of 1)~~
~~Allowable Positive MTC Limit~~

CTS

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

3.1.3 Moderator Temperature Coefficient (MTC) ~~(Analog)~~ 1

3.1.1.4 LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum positive limit shall be: ~~that specified in Figure 3.1.3-1.~~

DOC A02

a. +5 pcm/°F with THERMAL POWER < 70% RTP, and

b. +5 pcm/°F with THERMAL POWER at 70% RTP with a linear decrease to 0 pcm/°F at 100% RTP. 2

Applicability APPLICABILITY: MODES 1 and 2.
DOC M01
DOC A03

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limits.	A.1 Be in MODE 3.	6 hours

Action

SURVEILLANCE REQUIREMENTS

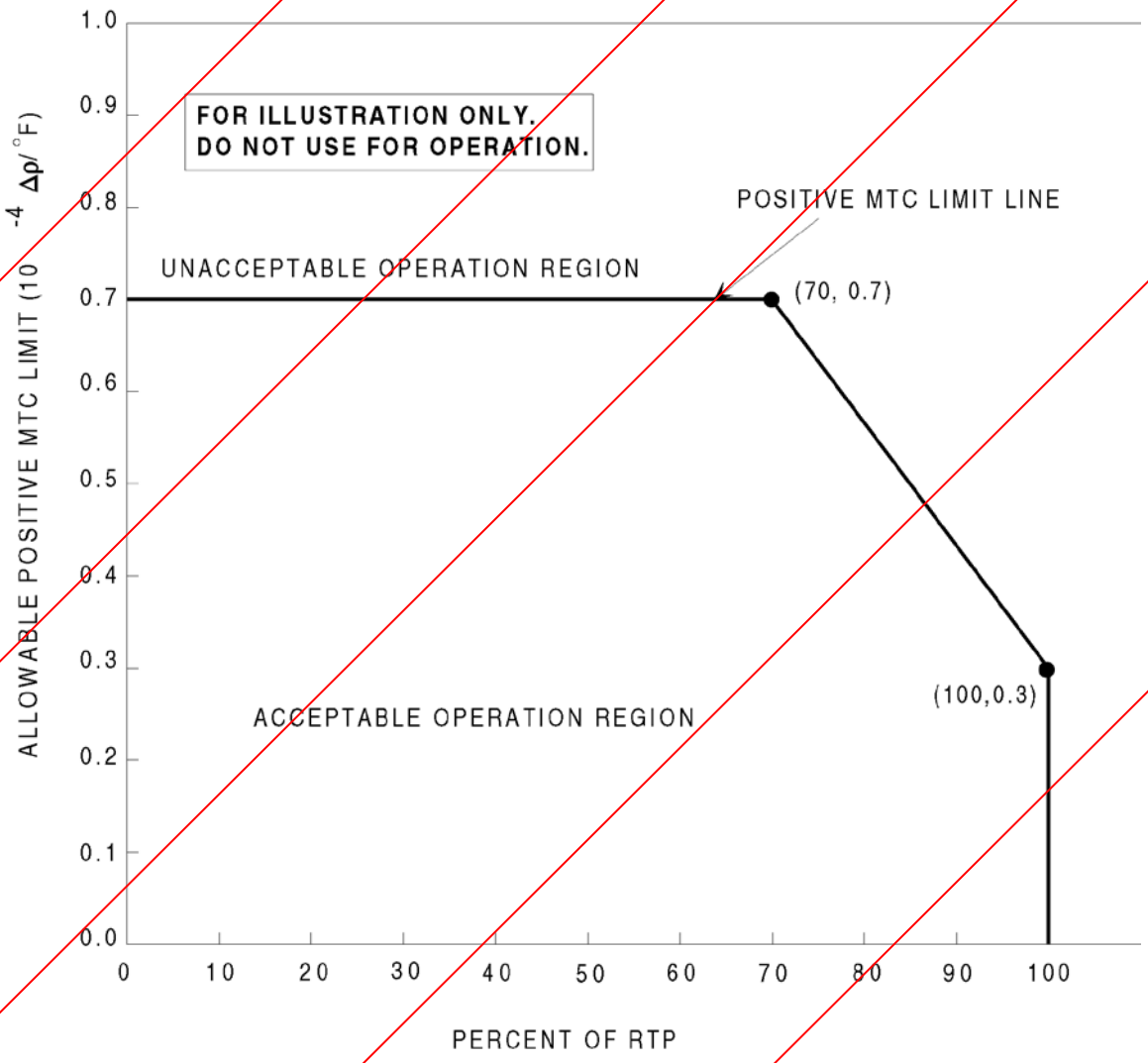
	SURVEILLANCE		FREQUENCY
4.1.1.4.1 4.1.1.4.1.a	SR 3.1.3.1	Verify MTC is within the upper limit specified in the COLR.	Prior to entering MODE 1 after each fuel loading ~~[~~ AND 4 --------NOTE-------- **Note Only required to be performed when MTC determined prior to entering MODE 1 is verified using adjusted predicted MTC ---------- 4.1.1.4.1.b Each fuel cycle within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	----------NOTES----------	5
***Note	If ~~the~~ MTC is more negative than the limit specified in the COLR when extrapolated to the end of cycle, SR 3.1.3.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit.	

4.1.1.4.2	Verify MTC is within the lower limit specified in the COLR.	~~Each fuel cycle within 7 EFPD of reaching 40 EFPD core burnup~~ 6 ~~AND~~
****Note DOC A04	1. Only required to be performed if MTC determined during performance of SR 3.1.3.1 is not within 1.6 pcm/°F of predicted MTC. 2.	Each fuel cycle within 7 EFPD of reaching 2/3 of expected core burnup 5

2



~~Figure 3.1.3-1 (page 1 of 1)~~
~~Allowable Positive MTC Limit~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

1. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. ISTS 3.1.3 contains Figure 3.1.3-1 for Allowable Positive MTC Vs Percent of RTP. This figure is not maintained in ITS 3.1.3. ITS 3.1.3 lists the maximum upper limit value based on THERMAL POWER level in the LCO. Therefore, ISTS Figure 3.1.3-1 is not required and has been deleted.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

5. ISTS SR 3.1.3.2 contains one Note. ITS SR 3.1.3.2 retains the Note but also adds a Note (Note 1) that states, "Only required to be performed if MTC determined during performance of SR 3.1.3.1 is not within 1.6 pcm/°F of predicted MTC." This Note is consistent with CTS and was approved in License Amendments 235 and 185, dated September 19, 2016, for PSL Units 1 and 2, respectively, as supported by Combustion Engineering Owners Group Topical Report CE NPSD-911-A and Amendment 1-A. The Note waives performance of the MTC end of cycle measurement if the measured MTC during performance of SR 3.1.3.1 is within 1.6 pcm/°F of the predicted value (NRC ADAMS Accession No. ML16183A138). Concomitant changes are made to reflect two Notes in the ITS.

6. The first Frequency of ISTS SR 3.1.3.2 states, "Each fuel cycle within 7 EFPD of reaching 40 EFPD core burnup." In September 2014, PSL modified the MTC surveillance requirements associated with the implementation of Westinghouse Topical Report WCAP-16011-P-A, "Startup Test Activity Reduction (STAR) Program." In the license amendment request supporting the change, PSL stated that proposed Surveillance 4.1.1.4.2 does not include a requirement for MTC measurements within 7 EFPD of reaching 40 EFPD core burnup to be extrapolated for comparison to the lower MTC limit because performing the surveillance at 2/3 of core burnup will be performed and available for extrapolation to verify the lower MTC limit. The extrapolation of MTC measured within 7 EFPD of reaching 40 EFPD core burnup will not add any additional technical value for comparison to the MTC lower limit and therefore the elimination of this frequency is considered acceptable. CTS 4.1.1.4.2, without the Frequency of 7 EFPD of reaching 40 EFPD core burnup, was approved in License Amendments 219 and 168, dated September 16, 2014, for PSL Units 1 and 2, respectively (NRC ADAMS Accession No. ML14218A180). Therefore, ITS SR 3.1.3.2 does not include this Frequency consistent with CTS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

B 3.1.3 Moderator Temperature Coefficient (MTC) ~~(Analog)~~

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended or rapid reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature. A positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less positive than that allowed by the LCO. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons (lumped burnable poison assemblies) to yield an MTC at the BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2) and

b. The MTC must be such that inherently stable power operations result during normal operation and during accidents, such as overheating and overcooling events.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations, to ensure the accident results are bounding ~~(Ref. 3)~~. 2

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the control element assembly (CEA) withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to a positive MTC is ~~a~~ an uncontrolled CEA withdrawal accident from zero power~~, also referred to as a startup accident~~ (Ref. ~~4~~). 2 2

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power is produced with all CEAs inserted, except the most reactive one, which is assumed withdrawn. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. A middle of cycle (MOC) measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

The MTC satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 2

BASES

LCO

LCO 3.1.3 requires the MTC to be within the positive and negative limits specified in the COLR to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The positive MTC limit in the COLR ensures that core overheating accidents will not violate the accident analysis assumptions. The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.

the LCO

The MTC limit in ~~Figure 3.1.3-1~~ is the maximum positive MTC value approved in the plant's licensing basis and ensures that the reactor operates with a negative MTC over the largest possible range of fuel cycle operation. The cycle-specific MTC limit specified in the COLR must be equal to or less positive than the MTC limit specified in ~~Figure 3.1.3-1~~.

the LCO

3

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed. Limited control of MTC can be achieved by adjusting CEA position and boron concentration. During operation the LCO can be ensured through measurement and adjustments to CEA position and boron concentration. The surveillance checks at BOC and MOC on an MTC provide confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on the MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure startup and subcritical accidents, such as the uncontrolled CEA or group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of the MTC, with temperature in MODES 3, 4, and 5, for DBAs initiated in MODES 1 and 2, is accounted for in the subject accident analysis. The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC and MOC measurements are used for normalization.

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The

BASES

ACTIONS (continued)

associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 and SR 3.1.3.2

The SRs for measurement of the MTC at the beginning and middle of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced to compensate for fuel depletion.

~~---------- REVIEWER'S NOTE ----------~~
~~The following Bases and the second Frequency of SR 3.1.3.1 are only applicable to plants that adopt WCAP-16011 (Reference 5).~~
~~----------~~ 4

[For fuel cycles that meet the applicability requirements in Reference ~~5~~, and specifically the acceptance criteria that must be met in order to substitute the measured value of MTC at hot zero power (HZP) with an alternate MTC value, SR 3.1.3.1 may be met prior to entering MODE 1 after each fuel loading by confirmation that the predicted MTC, when adjusted for the measured RCS boron concentration, is within the most positive (least negative) MTC limit specified in the COLR. If this adjusted predicted MTC value is used to meet the SR prior to entering MODE 1, a confirmation by measurement that MTC is within the upper MTC limit must be performed in MODE 1 within 7 Effective Full Power Days (EFPD) after reaching 40 EFPD of core burnup. The applicability requirements in Reference ~~5~~ ensure core designs are not significantly different from those used to benchmark predictions and require that the measured RCS boron concentration meets specific test criteria. This provides assurance that the MTC obtained from the adjusted predicted MTC is accurate. 5 3 3 2 3 2

For fuel cycles that do not meet the applicability requirements in Reference ~~5~~, the verification of MTC required prior to entering MODE 1 after each fuel loading is performed by calculation of the MTC based on measurement of the isothermal temperature coefficient. In this case, measurement of MTC within 7 EFPD after reaching 40 EFPD of core burnup is not required.] 3 2 5 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

The requirement for measurement prior to operation > 5% RTP satisfies the confirmatory check on the most positive (least negative) MTC value.

The requirement for measurement, within 7 EFPD after reaching ~~40 EFPD at and~~ 2/3 core burnup, satisfies the confirmatory check of the most negative MTC value. The measurement is performed at any THERMAL POWER, so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values may be extrapolated and compensated to permit direct comparison to the MTC limits specified in the COLR. 3

Insert 1 2 3

SR 3.1.3.2 is modified by ~~a Note, which~~ indicates that if the extrapolated MTC is more negative than the EOC limit specified in the COLR, the Surveillance may be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. An engineering evaluation is performed if the extrapolated value of MTC exceeds the Specification limits.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. FSAR, ~~Section [ ]~~. Chapter 15 U
~~3. FSAR, Section [ ].~~
~~4. FSAR, Section [ ].~~
3 ~~5~~. ~~[~~WCAP-16011-P-A, Rev. 0, "Startup Test Activity Reduction Program," dated February 2005.~~]~~ 2 5 3
4. Combustion Engineering Owners Group Topical Report CE NPSD-911-A and Amendment 1-A, “Analysis of Moderator Temperature Coefficients in Support of a Change in the Technical Specifications End-Of-Cycle Negative MTC Limit,” dated June 2000. 2 3

INSERT 1 (2) (3)

two Notes. Note 1 waives performance of the MTC EOC verification if the MTC determined during performance of SR 3.1.3.1 is within 1.6 pcm/°F of the predicted value. This allowance is based on the methodology described in Topical Report CE NPSD-911-A and Amendment 1-A (Ref. 4). Each cycle, the core is designed using this CE methodology such that the best estimate MTC includes a design margin to the BOC and EOC limits. The design margin is determined to be 1.6 pcm/°F at all times in core life. Therefore, if the BOC MTC values are within 1.6 pcm/°F of the best estimate prediction, then it can be assumed that the EOC MTC values will also be within 1.6 pcm/°F of the prediction and its verification is not required. Note 2

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

B 3.1.3 Moderator Temperature Coefficient (MTC) ~~(Analog)~~

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended or rapid reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature. A positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less positive than that allowed by the LCO. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons (lumped burnable poison assemblies) to yield an MTC at the BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2) and

b. The MTC must be such that inherently stable power operations result during normal operation and during accidents, such as overheating and overcooling events.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations, to ensure the accident results are bounding ~~(Ref. 3)~~. 2

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the control element assembly (CEA) withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to a positive MTC is ~~a~~ CEA withdrawal accident from zero power~~, also referred to as a startup accident~~ (Ref. ~~4~~). 2

2 an uncontrolled

2

2

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power is produced with all CEAs inserted, except the most reactive one, which is assumed withdrawn. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. A middle of cycle (MOC) measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

The MTC satisfies Criterion 2 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii) 2

BASES

LCO

LCO 3.1.3 requires the MTC to be within the positive and negative limits specified in the COLR to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The positive MTC limit in the COLR ensures that core overheating accidents will not violate the accident analysis assumptions. The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.

the LCO

The MTC limit in ~~Figure 3.1.3-1~~ is the maximum positive MTC value approved in the plant's licensing basis and ensures that the reactor operates with a negative MTC over the largest possible range of fuel cycle operation. The cycle-specific MTC limit specified in the COLR must be equal to or less positive than the MTC limit specified in ~~Figure 3.1.3-1~~.

the LCO

3

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed. Limited control of MTC can be achieved by adjusting CEA position and boron concentration. During operation the LCO can be ensured through measurement and adjustments to CEA position and boron concentration. The surveillance checks at BOC and MOC on an MTC provide confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on the MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure startup and subcritical accidents, such as the uncontrolled CEA or group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of the MTC, with temperature in MODES 3, 4, and 5, for DBAs initiated in MODES 1 and 2, is accounted for in the subject accident analysis. The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC and MOC measurements are used for normalization.

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The

BASES

ACTIONS (continued)

associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 and SR 3.1.3.2

The SRs for measurement of the MTC at the beginning and middle of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced to compensate for fuel depletion.

~~---------- REVIEWER'S NOTE ----------~~
~~The following Bases and the second Frequency of SR 3.1.3.1 are only applicable to plants that adopt WCAP-16011 (Reference 5).~~
~~----------~~ 4

[For fuel cycles that meet the applicability requirements in Reference ~~5~~, and specifically the acceptance criteria that must be met in order to substitute the measured value of MTC at hot zero power (HZP) with an alternate MTC value, SR 3.1.3.1 may be met prior to entering MODE 1 after each fuel loading by confirmation that the predicted MTC, when adjusted for the measured RCS boron concentration, is within the most positive (least negative) MTC limit specified in the COLR. If this adjusted predicted MTC value is used to meet the SR prior to entering MODE 1, a confirmation by measurement that MTC is within the upper MTC limit must be performed in MODE 1 within 7 Effective Full Power Days (EFPD) after reaching 40 EFPD of core burnup. The applicability requirements in Reference ~~5~~ ensure core designs are not significantly different from those used to benchmark predictions and require that the measured RCS boron concentration meets specific test criteria. This provides assurance that the MTC obtained from the adjusted predicted MTC is accurate. 5 3 3 2 3 2

For fuel cycles that do not meet the applicability requirements in Reference ~~5~~, the verification of MTC required prior to entering MODE 1 after each fuel loading is performed by calculation of the MTC based on measurement of the isothermal temperature coefficient. In this case, measurement of MTC within 7 EFPD after reaching 40 EFPD of core burnup is not required.] 3 2 5 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

The requirement for measurement prior to operation > 5% RTP satisfies the confirmatory check on the most positive (least negative) MTC value.

The requirement for measurement, within 7 EFPD after reaching ~~40 EFPD at and~~ 2/3 core burnup, satisfies the confirmatory check of the most negative MTC value. The measurement is performed at any THERMAL POWER, so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values may be extrapolated and compensated to permit direct comparison to the MTC limits specified in the COLR. 3

Insert 1 2 3

SR 3.1.3.2 is modified by ~~a Note, which~~ indicates that if the extrapolated MTC is more negative than the EOC limit specified in the COLR, the Surveillance may be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. An engineering evaluation is performed if the extrapolated value of MTC exceeds the Specification limits.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. FSAR, ~~Section [ ]~~. Chapter 15

U

~~3. FSAR, Section [ ].~~

~~4. FSAR, Section [ ].~~

3

~~5~~. ~~[~~WCAP-16011-P-A, Rev. 0, "Startup Test Activity Reduction Program," dated February 2005.~~]~~

2 5

3

4. Combustion Engineering Owners Group Topical Report CE NPSD-911-A and Amendment 1-A, “Analysis of Moderator Temperature Coefficients in Support of a Change in the Technical Specifications End-Of-Cycle Negative MTC Limit,” dated June 2000.

2 3

INSERT 1 (2) (3)

two Notes. Note 1 waives performance of the MTC EOC verification if the MTC determined during performance of SR 3.1.3.1 is within 1.6 pcm/°F of the predicted value. This allowance is based on the methodology described in Topical Report CE NPSD-911-A and Amendment 1-A (Ref. 4). Each cycle, the core is designed using this CE methodology such that the best estimate MTC includes a design margin to the BOC and EOC limits. The design margin is determined to be 1.6 pcm/°F at all times in core life. Therefore, if the BOC MTC values are within 1.6 pcm/°F of the best estimate prediction, then it can be assumed that the EOC MTC values will also be within 1.6 pcm/°F of the prediction and its verification is not required. Note 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.3 BASES, MODERATOR TEMPERATURE COEFFICIENT (MTC)

1. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. Changes are made to be consistent with changes made to the Specification.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 4

3.1.4 Control Element Assembly (CEA) Alignment

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTIVITY CONTROL SYSTEMS

~~3/4.1.3 MOVABLE CONTROL ASSEMBLIES~~

Control Element Assembly ( ) Alignment

~~FULL LENGTH~~ CEA ~~POSITION~~

LIMITING CONDITION FOR OPERATION

LCO 3.1.4

~~3.1.3.1~~ , and The CEA Block Circuit and all ~~full length (shutdown and regulating)~~ CEAs shall be OPERABLE ~~with each~~ CEA ~~of a given group positioned~~ within 7.5 inches (indicated position) of ~~all other~~ ~~CEAs in its~~ group~~.~~ the CEA deviation circuit shall be OPERABLE. shall be aligned to AND all s their respective

M01 A02

Applicability

APPLICABILITY: MODES 1~~*~~ and 2~~*~~.

A03

ACTION:

ACTION F

~~a.~~ ~~With~~ one or more ~~full length~~ CEAs inoperable ~~due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable,~~ ~~determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour~~ and be in ~~HOT~~ STANDBY within 6 hours.

A02 LA01 L01

Required Action F.1

MODE 3

~~b.~~ ~~With the~~ CEA Block Circuit inoperable, within 6 hours ~~either:~~

~~1. With one CEA position indicator per group inoperable, take action per Specification 3.1.3.3, or~~

A04

Required Action D.2.2. NOTE

NOTE Required Action D.2.2 shall not be performed when in conflict with either Required Action C.1, C.2, or E.1.

L02

Fully withdraw all

~~2. With the group overlap and/or sequencing interlocks inoperable, maintain~~ CEAs in groups 3, 4, 5 and 6 ~~fully withdrawn~~ and withdraw ~~the~~ CEAs in group 7 to less than 5% insertion and place and maintain the CEA drive ~~system mode~~ switch in either the "Manual" or "Off" position~~, or~~

Required Action D.2.2

all

Required Action F.1

3. Be in ~~at least~~ ~~HOT~~ STANDBY.

MODE 3

~~c. With one full length CEA inoperable due to causes other than addressed by Action a above, but within its above specified alignment requirements and either fully withdrawn or within the long term steady state insertion limits if in CEA group 7, operation in MODES 1 and 2 may continue.~~

A05

ACTION A Condition A

~~d.~~ ~~With~~ one or more ~~full length~~ CEAs misaligned from ~~any other CEAs in~~ its group by more than 7.5 inches but less than 15 inches~~, operation in MODES 1 and 2 may continue, provided that~~ within ~~one hour the misaligned CEA(s) is either:~~

A02

2 hours

Required Action A.1

1. Restore~~d~~ to OPERABLE status within its above specified alignment ~~requirements, or~~

CEA within 2 hours

A06

~~* See Special Test Exceptions 3.10.2 and 3.10.5.~~

A03

REACTIVITY CONTROL SYSTEMS

Control Element Assembly ( ) Alignment

~~FULL LENGTH~~ CEA ~~POSITION (continued)~~

LIMITING CONDITION FOR OPERATION (continued)

~~2. Declared inoperable and satisfy SHUTDOWN MARGIN requirements of Specification 3.1.1.1. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year provided all of the following conditions are met:~~ L02 A07

Required Action A.1 — 2 hours

~~a)~~ Within ~~1 hour~~, ~~the remainder of the CEAs in the group with the inoperable~~ CEA shall be aligned to within 7.5 inches ~~of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.~~ A06

~~b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.~~ L01

Required Action F.1 — in MODE 3

~~Otherwise,~~ be ~~at least HOT STANDBY~~ within the next 6 hours.

ACTION B Condition B

~~e.~~ ~~With~~ one ~~full length~~ CEA misaligned from ~~any other CEA in~~ its group by 15 or more inches~~, operation in MODES 1 and 2 may continue provided that~~ the misaligned CEA is positioned within 7.5 inches of other CEAs in its group in accordance with the ~~time constraints shown in~~ COLR ~~Figure 3.1-1a~~. A02

Required Action B.1 — LA02

ACTION C Condition C

~~f.~~ ~~With~~ one ~~full length~~ CEA misaligned from ~~any other CEA in~~ its group by 15 or more inches beyond the time ~~constraints shown~~ in COLR ~~Figure 3.1-1a,~~ reduce ~~power~~ to ≤ 70% ~~of RATED~~ THERMAL POWER prior to completing ACTION f.1 or f.2. A02

Required Action C.1 — RTP within 1 hour. M02

Required Action C.2

1. Restore~~d the~~ CEA ~~to OPERABLE status within its specified~~ alignment ~~requirements, or~~ — within 2 hours M02

~~2. Declare the CEA inoperable and satisfy the SHUTDOWN MARGIN requirements of Specification 3.1.1.1. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:~~ L01

Required Action C.2 — 2 hours

~~a)~~ Within ~~1 hour~~, ~~the remainder of the CEAs in the group with~~ the inoperable CEA shall be aligned to within 7.5 inches ~~of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.~~ A06

REACTIVITY CONTROL SYSTEMS

Control Element Assembly ( ) Alignment

~~FULL LENGTH~~ CEA ~~POSITION (continued)~~

LIMITING CONDITION FOR OPERATION (continued)

~~b)~~ ~~The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.~~ L01

Required Action F.1

~~Otherwise,~~ be ~~at least HOT STANDBY~~ within the next 6 hours.

in MODE 3 Two or more s

Condition F

Required Action F.1

~~g.~~ ~~With more than one full length~~ CEA ~~inoperable or~~ misaligned ~~from any other CEA in its group~~ by 15 inches ~~(indicated position) or more~~, be in ~~HOT STANDBY~~ within 6 hours. A02

≥ MODE 3

~~h.~~ ~~With one full length CEA inoperable due to causes other than addressed by ACTION a above, and inserted beyond the long term steady state insertion limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.~~ A05

SURVEILLANCE REQUIREMENTS

Verify indicated

SR 3.1.4.1 ~~4.1.3.1.1~~ The position of each ~~full length~~ CEA ~~shall be determined~~ to be within 7.5 inches ~~(indicated position)~~ of all other CEAs in its group in accordance with the Surveillance Frequency Control Program ~~except during time intervals when the~~ Deviation Circuit and/or CEA Block Circuit ~~are~~ inoperable, then verify the individual CEA positions ~~at least once per~~ 4 hours. thereafter A02

Required Action D.1, E.1 Completion Time

1 hour AND Every M03

Verify CEA freedom of movement (trippability) by moving

SR 3.1.4.4 ~~4.1.3.1.2~~ Each ~~full length~~ CEA not fully inserted ~~shall be determined to be OPERABLE by inserting it at least~~ 7.5 inches in accordance with the Surveillance Frequency Control Program. A02

individual

Verify in either direction that is

SR 3.1.4.2 ~~4.1.3.1.3~~ The CEA Block Circuit ~~shall be demonstrated~~ is OPERABLE in accordance with the Surveillance Frequency Control Program ~~by a functional test which verifies that the circuit prevents any CEA from being misaligned from all other CEAs in its group by more than 7.5 inches (indicated position).~~ LA03

Verify is

SR 3.1.4.2 ~~4.1.3.1.4~~ The CEA Block Circuit ~~shall be demonstrated~~ OPERABLE ~~by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit prevents the regulating CEAs from being inserted beyond the Power Dependent Insertion Limit of COLR Figure 3.1-2:~~ LA03

~~*a. Prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 92 days, and~~ A08

~~b.~~ In accordance with the Surveillance Frequency Control Program.

SR 3.1.4.3 Verify the CEA deviation circuit is OPERABLE in accordance with the Surveillance Frequency Control Program. M01

~~* The licensee shall be excepted from compliance during the startup test program for an entry into MODE 2 from MODE 3 made in association with a measurement of power defect.~~ A08

~~DELETED~~

REACTIVITY CONTROL SYSTEMS

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

SR 3.1.4.5

3.1.3.4 ~~The individual full length (shutdown and control)~~ Verify each CEA drop time is ~~, from a fully withdrawn position, shall be~~ ≤ 3.1 seconds ~~from when electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with~~: LA04

~~a. T_{avg} ≥ 515°F, and~~

~~b. All reactor coolant pumps operating.~~

APPLICABILITY: ~~MODE 3.~~ A09

ACTION:

~~a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.~~ A09

~~b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.~~ A10

SURVEILLANCE REQUIREMENTS

SR 3.1.4.5

~~4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement~~ prior to reactor criticality~~:~~

~~a. For all CEAs following~~ after each removal of the reactor vessel head~~,~~

~~b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and~~ L03

~~c. In accordance with the Surveillance Frequency Control Program.~~ L04

A01

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

See ITS 3.1.1

SHUTDOWN MARGIN - T_{avg} > 200 °F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODES 1, 2*, 3 and 4.

See ITS 3.1.6

ACTION:

With the SHUTDOWN MARGIN not within limits immediately initiate and continue boration at ≥ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

L01

a. ~~Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable~~. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. When in MODES 1 or 2#, in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

See ITS 3.1.6

c. When in MODE 2## at least once during CEA withdrawal and in accordance with the Surveillance Frequency Control Program until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

See ITS 3.1.1

* See Special Test Exception 3.10.1.

\# With K_{eff} ≥ 1.0.

With K_{eff} < 1.0.

See ITS 3.1.6

REACTIVITY CONTROL SYSTEMS

A01

SHUTDOWN MARGIN - T_{avg} ≤ 200 °F

See ITS 3.1.1

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

Within the limits specified in the COLR, and in addition with the Reactor Coolant System drained below the hot leg centerline, one charging pump shall be rendered inoperable.*

See ITS 3.4.12

APPLICABILITY: MODE 5.

ACTION:

See ITS 3.1.1

If the SHUTDOWN MARGIN requirements cannot be met, immediately initiate and continue boration at ≥ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN requirements of Specification 3.1.1.2 shall be determined:

a. ~~Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable.~~

L01

If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

See ITS 3.1.1

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2.b and by verifying at least one charging pump is rendered inoperable.*

See ITS 3.4.12

* Breaker racked-out.

See ITS 3.4.12

A01

REACTIVITY CONTROL SYSTEMS

Control Element Assembly

~~3/4.1.3 MOVABLE CONTROL ASSEMBLIES~~

( ) Alignment

CEA ~~POSITION~~

LIMITING CONDITION FOR OPERATION

M01

A02

LCO 3.1.4

, and

~~3.1.3.1~~ The CEA Block Circuit and all ~~full-length (shutdown and regulating)~~ CEAs ~~which are inserted in the core,~~ shall be OPERABLE ~~with each~~ CEA ~~of a given group positioned~~ within 7.0 inches (indicated position) of ~~all other~~ CEAs ~~in its~~ group.

the CEA deviation circuit shall be OPERABLE.

shall be aligned to

AND all

s

their respective

Applicability

APPLICABILITY: MODES 1~~*~~ and 2~~*~~.

A03

ACTION:

A02

LA01

ACTION F

~~a.~~ ~~With~~ one or more ~~full-length~~ CEAs inoperable ~~due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable,~~ determine that ~~the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour~~ and be in at least ~~HOT STANDBY~~ within 6 hours.

MODE 3

Required Action F.1

L01

~~b.~~ ~~With the~~ CEA Block Circuit inoperable, within 6 hours ~~either:~~

A04

~~1. With one CEA position indicator per group inoperable take action per Specification 3.1.3.2, or~~

Required Action D.2.2. NOTE

NOTE Required Action D.2.2 shall not be performed when in conflict with either Required Action C.1, C.2, or E.1.

L02

Fully withdraw all

~~2.~~ ~~With the group overlap and/or sequencing interlocks inoperable maintain~~ CEA groups 1, 2, 3 and 4 ~~fully withdrawn~~ and ~~the~~ CEAs in group 5 to less than 15% insertion and place and maintain the CEA drive ~~system~~ in either the "Manual" or "Off" position~~, or~~

withdraw all

Required Action D.2.2

Required Action F.1

~~3.~~ Be in ~~at least HOT STANDBY~~.

MODE 3

Two or more

s

A02

Condition F

~~c.~~ ~~With more than one full-length~~ CEA ~~inoperable or~~ misaligned ~~from any other CEA in its group~~ by more than 15 inches ~~(indicated position),~~ be in ~~at least HOT STANDBY~~ within 6 hours.

Required Action F.1

MODE 3

A02

ACTION B Condition B

~~d.~~ ~~With~~ one ~~full-length~~ CEA misaligned from ~~any other CEA in~~ its group by more than 15 inches~~, operation in MODES 1 and 2 may continue, provided that the misaligned~~ CEA is positioned within 15 inches of the other CEAs in its group ~~in accordance with the time constraints shown~~ in COLR ~~Figure 3.1-1a.~~

Required Action B.1

LA02

~~* See Special Test Exceptions 3.10.2, 3.10.4 and 3.10.5.~~

A03

A01

REACTIVITY CONTROL SYSTEMS

ACTION: (Continued)

LA02 L01

ACTION C
Condition C
Required Action
C.1

e. ~~With~~ one ~~full-length~~ CEA misaligned from ~~any other CEA in~~ its group by ~~more than~~ 15 inches beyond the time ~~constraints shown~~ in COLR ~~Figure 3-1-1a,~~ reduce ~~power~~ to ≤ 70% ~~of RATED~~ THERMAL POWER prior to completing ACTION e.1 or e.2.

A02 M02

> RTP within 1 hour.

Required Action
C.2

1. Restore ~~the~~ CEA ~~to OPERABLE status within its specified~~ alignment ~~requirements, or~~

M02

within 2 hours

2. ~~Declare the CEA inoperable and satisfy SHUTDOWN MARGIN requirement of Specification 3.1.1.1. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:*~~

A06

Required Action
C.2

2 hours

a) Within ~~1 hour~~ the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.0 inches ~~of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.~~

b) ~~The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.~~

L01

Required Action F.1

~~Otherwise,~~ be in ~~at least HOT STANDBY~~ within the next 6 hours.

in MODE 3

ACTION A
Condition A

f. ~~With~~ one or more ~~full-length~~ CEA(s) misaligned from ~~any other CEAs in~~ its group by more than 7.0 inches but less than or equal to 15 inches ~~, operation in MODES 1 and 2 may continue, provided that~~ within ~~1 hour the misaligned CEA(s) is either~~:

A02

2 hours

Required Action
A.1

1. Restore ~~d to OPERABLE status within its above specified~~ alignment ~~requirements, or~~

A06

CEA within 2 hours

2. ~~Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is~~ satisfied. ~~After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:~~

A06

2 hours

Required Action
A.1

a) Within ~~1 hour~~ the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.0 inches ~~of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.~~

b) ~~The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.~~

L01

Required Action F.1

~~Otherwise,~~ be in ~~at least HOT STANDBY~~ within the next 6 hours.

in MODE 3

g. ~~With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.~~

A05

~~* If the pre-misalignment ASI was more negative than -0.15, reduce power to ≤ 70% of RATED THERMAL POWER or 70% of the THERMAL POWER level prior to the misalignment, whichever is less, prior to completing ACTION e.2.a) and e.2.b).~~

L05

REACTIVITY CONTROL SYSTEMS

ACTION: (Continued)

~~h. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, but within its above specified alignment requirements and either fully withdrawn or within the Long Term Steady State Insertion Limits if in full-length CEA group 5, operation in MODES 1 and 2 may continue.~~ A05

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 ~~4.1.3.1.1~~ Verify The indicated Position of each ~~full-length~~ CEA ~~shall be determined~~ to be within 7.0 inches ~~(indicated position)~~ of all other CEAs in its group in accordance with the Surveillance Frequency Control Program ~~except during time intervals when the~~ Deviation Circuit and/or CEA Block Circuit ~~are~~ inoperable, then verify the individual CEA positions ~~at least once per~~ 4 hours. thereafter A02

Required Action D.1, E.1 Completion Time

SR 3.1.4.4 ~~4.1.3.1.2~~ 1 hour AND Every Verify CEA freedom of movement (trippability) by moving Each ~~full-length~~ CEA not fully inserted ~~in the core shall be determined to be OPERABLE by movement of at least~~ 7.0 inches in ~~any one~~ direction in accordance with the Surveillance Frequency Control Program. M03 A02

individual that is either

SR 3.1.4.2 ~~4.1.3.1.3~~ Verify The CEA Block Circuit ~~shall be demonstrated~~ is OPERABLE in accordance with the Surveillance Frequency Control Program ~~by a functional test which verifies that the circuit prevents any CEA from being misaligned from all other CEAs in its group by more than 7.0 inches (indicated position).~~ LA03

SR 3.1.4.2 ~~4.1.3.1.4~~ Verify The CEA Block Circuit ~~shall be demonstrated~~ is OPERABLE ~~by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit prevents the regulating CEAs from being inserted beyond the Power Dependent Insertion Limit of COLR Figure 3.1-2:~~ LA03

~~*a. Prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 92 days, and~~ A08

~~b.~~ In accordance with the Surveillance Frequency Control Program.

SR 3.1.4.3 Verify the CEA deviation circuit is OPERABLE in accordance with the Surveillance Frequency Control Program.. M01

~~* The licensee shall be excepted from compliance during the initial startup test program for an entry into MODE 2 from MODE 3 made in association with a measurement of power defect.~~ A08

REACTIVITY CONTROL SYSTEMS

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

SR 3.1.4.5

~~3.1.3.4 The individual full-length (shutdown and regulating)~~ Verify each CEA drop time ~~from a fully withdrawn position, shall be less than or equal to~~ is ≤ 3.25 seconds ~~from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90% insertion position with:~~ LA04

~~a. T_{avg} greater than or equal to 515°F, and~~

~~b. All reactor coolant pumps operating.~~

APPLICABILITY: MODES 1 and 2.

ACTION:

ITS 3.1.4 Condition F

~~a. With the drop time of any full-length CEA determined to exceed the above limit:~~ One or more CEAs inoperable

ITS 3.1.4 Required Action F.1

~~1. If in MODE 1 or 2,~~ be in at least HOT STANDBY within 6 hours, ~~or~~

~~2. If in MODE 3, 4, or 5, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.~~ A09

~~b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.~~ A10

SURVEILLANCE REQUIREMENTS

SR 3.1.4.5

~~4.1.3.4 The CEA drop time of full-length CEAs shall be demonstrated through measurement~~ prior to reactor criticality~~:~~

~~a. For all CEAs following~~ after each removal ~~and installation~~ of the reactor vessel head~~,~~

~~b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and~~ L03

~~c. In accordance with the Surveillance Frequency Control Program.~~ L04

A01

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

See ITS 3.1.1

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODES 1, 2*, 3 and 4.

See ITS 3.1.6

ACTION:

With the SHUTDOWN MARGIN outside the COLR limits, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

L01

a. ~~Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable.~~ If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

See ITS 3.1.6

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

See ITS 3.1.6

A01

ITS 3.1.4

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODE 5.

See ITS 3.1.1

ACTION:

With the SHUTDOWN MARGIN outside the COLR limits, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

a. ~~Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable.~~ If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

L01

See ITS 1.1

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

See ITS 3.1.1

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2b and by verifying at least two charging pumps are rendered inoperable by racking out their motor circuit breakers.

See ITS 3.4.12

DISCUSSION OF CHANGES
ITS 3.1.4, CONTROL ELEMENT ASSEMBLY (CEA) ALIGNMENT

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS). These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.1.3.1 LCO, Actions a, d, e, f, g, 4.1.3.1.1, and 4.1.3.1.2 include the descriptor “full length” in reference to CEA(s). Unit 2 CTS 3.1.3.1 LCO, Actions a, c, d, e, f, 4.1.3.1.1, and 4.1.3.1.2 include the descriptor “full length” in reference to CEA(s).

The purpose of the CTS 3.1.3.1 descriptor is to distinguish between full length and partial length CEA(s). Unit 1 and Unit 2 no longer use partial length CEA(s). Therefore, since partial length CEA(s) are no longer a design feature for Unit 1 and Unit 2, the “full length” descriptors used for CEA(s) are deleted. These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A03 Unit 1 CTS 3.1.3.1 Applicability is MODES 1 and 2 with a footnote (footnote *) for MODE 1 and MODE 2 stating "See Special Test Exception 3.10.2 and 3.10.5.” Unit 2 CTS 3.1.3.1 Applicability is MODES 1 and 2 with a footnote (footnote *) for MODE 1 and MODE 2 stating "See Special Test Exception 3.10.2, 3.10.4, and 3.10.5.” ITS 3.1.4 does not contain the footnote or a reference to the Special Test Exceptions. This changes the CTS by not including footnote * in the ITS.

The purpose of the footnote references is to alert the user that conditions exist that may modify the Applicability of the Specification. ITS LCO 3.0.7 provides guidance on the use of special test exceptions. It is an ITS convention to not include reference only type footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

A04 Unit 1 CTS 3.1.3.1 ACTION b.1 states, in part, that “With one CEA position indicator per group inoperable, take action per Specification 3.1.3.3.” Unit 2 CTS 3.1.3.1 ACTION b.1 states, in part, that “With one CEA position indicator per group inoperable, take action per Specification 3.1.3.2.” ITS 3.1.4 does not provide a reference to ITS 3.1.6. This changes the CTS by deleting the reference to another Specification.

The purpose of the CTS 3.1.3.1 statement, “with one CEA position indicator per group inoperable, take action per Specification…” is to provide a reference to another Specification that may apply under these conditions. This changes the CTS by not including the cross-reference in the ITS. It is an ITS convention to not include footnotes or cross-references to comply with other Specification LCO and actions. CTS 3.0.1 (ITS LCO 3.0.1) requires LCOs to be met during the MODES or other specified conditions in the Applicability, except as provided in

LCO 3.0.2. CTS 3.0.2 (ITS LCO 3.0.2) requires, in part, upon discovery of a failure to meet an LCO, the Actions to be met. Therefore, it is unnecessary to reference or duplicate the requirements of another Specification. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

A05 Unit 1 CTS 3.1.3.1 ACTION h. and Unit 2 CTS ACTION g. each state that with one full length CEA inoperable due to causes other than addressed by ACTION a above, and inserted beyond the long term steady state insertion limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6. ITS 3.1.4 does not provide a reference to ITS 3.1.6 (CTS 3.1.3.6). This changes the CTS by deleting the reference to Specification 3.1.3.6.

Unit 1 CTS 3.1.3.1 ACTION c. and Unit 2 CTS ACTION h. each state that with one full length CEA inoperable due to causes other than addressed by Action a above, but within its above specified alignment requirements and either fully withdrawn or within the long term steady state insertion limits if in CEA group 7 (for Unit 1) and if in CEA group 5 (for Unit 2), operation in MODES 1 and 2 may continue. Though specific reference to other Specifications is not made, reference to Specifications 3.1.3.5 and 3.1.3.6 is implied and requirements from these Specifications and the COLR are duplicated within this statement. ITS 3.1.4 does not include duplicative requirements from other Specifications. This changes the CTS by deleting a statement that contains duplicative requirements from other Specifications.

The purpose of the CTS statements is to associate requirements from other Specifications that may apply under these conditions with a CEA that is "trippable." This changes the CTS by not including these statements in the ITS. It is an ITS convention to not include these type of references or duplicative requirements from other Specifications. CTS 3.0.1 (ITS LCO 3.0.1) requires LCOs to be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2. CTS 3.0.2 (ITS LCO 3.0.2) requires, in part, upon discovery of a failure to meet an LCO, the Actions to be met. Therefore, it is unnecessary to reference or duplicate the requirements of another Specification. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

A06 Unit 1 CTS 3.1.3.1 Action d, and Unit 2 Action f, each allow, when one or more CEAs are misaligned by more than the LCO limit, but < 15 inches for Unit 1 and ≤ 15 inches for Unit 2, operation to continue in MODES 1 and 2 provided within 1 hour action is taken per d.1 or d.2 (Unit 1) and f.1 or f.2 (Unit 2).

Unit 1 CTS 3.1.3.1 Actions d.2 and f.2, and Unit 2 Actions e.2 and f.2, each state that after declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within the LCO tolerance of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

The purpose of the CTS statement is to provide time to restore the CEA to within alignment limits. CTS 3.1.3.1 Unit 1 Action d, and Unit 2 Action f specifically allow one hour to either align the CEA (i.e., move the misaligned CEA to align with the CEA group) or declare the CEA inoperable and provide an additional hour to move the other CEAs within the group to align with the misaligned CEA while maintaining allowable CEA sequence and insertion limits specified in the referenced COLR figure and THERMAL POWER restriction specified in Specification 3.1.3.6. In effect, the CTS Actions provide a total of 2 hours to align a CEA to within limits in this condition. This changes CTS by combining the 1 hour to restore the CEA alignment and additional 1 hour to restore CEA alignment.

CTS 3.1.3.1 Unit 1 Action f, and Unit 2 Action e require, in part, THERMAL POWER to be reduced ≤ 70 RTP prior to performing Action f.2 (Unit 1) and Action e.2 (Unit 2). The time to reduce THERMAL POWER has been defined as one hour (See DOC M02). After the power reduction, Action f.2 (Unit 1) and Action e.2 (Unit 2) require declaring the CEA inoperable and provide an additional hour to move the other CEAs within the group to align with the misaligned CEA while maintaining allowable CEA sequence and insertion limits specified in the referenced COLR figure and THERMAL POWER restriction specified in Specification 3.1.3.6. In effect, the CTS Actions provide time to reduce THERMAL POWER (i.e., 1 hour per ITS 3.1.4, Required Action C.1) and an additional hour (i.e., total of 2 hours from Condition entry) to align a CEA to within limits in this condition. This changes CTS by combining the 1 hour to reduce THERMAL POWER and additional 1 hour to restore CEA alignment.

ITS Section 1.2 provides guidance on logical connectors and ITS Section 1.3 provides guidance on Completion Times. When two actions are connected by the logical connector “or” either action can be performed within the required Completion Time. Thus, a total of 2 hours is provided in ITS to align the CEA regardless of the method of restoring the alignment (i.e., moving the CEA to align or moving the CEA group to align).

Also, CTS 3.1.3.6 LCO includes reference to limits specified in the COLR that include the COLR figure specifying the CEA sequence and insertion limits and THERMAL POWER level. This changes the CTS by not including the reference to the COLR figure and Specification 3.1.3.6 in the ITS. It is an ITS convention to not include these type of references and duplicative requirements from other Specifications. CTS 3.0.1 (ITS LCO 3.0.1) requires LCOs to be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2. CTS 3.0.2 (ITS LCO 3.0.2) requires, in part, upon discovery of a failure to meet an LCO, the ACTIONS to be met. Therefore, it is unnecessary to reference or duplicate the requirements of another Specification.

This change is designated as administrative as it combines the 1 hour and additional 1 hour to restore CEA alignment into a single 2 hour Completion Time and removes reference to other Specification requirements with no technical change to the CTS.

A07 **Unit 1 only:** CTS 3.1.3.1 ACTION d.2 states that with one or more CEAs misaligned from any other CEAs in its group by more than 7.5 inches but less than 15 inches, after declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 "for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year." Additionally, CTS 3.1.3.1 ACTION d.2.a) requires that within 1 hour, the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.5 inches of the inoperable CEA (i.e., within 1 hour, meet the LCO requirements of CTS 3.1.3.1). ITS 3.1.4 ACTIONS do not include the operational limit "for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year." This changes the CTS by deleting the operational limit "for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year." ITS 3.1.4 retains appropriate CTS 3.1.3.1 actions associated with a CEA outside alignment limits, as modified per the Discussion of Changes provided herein.

The purpose of the CTS requirement is to limit the time to operate, once CEA alignment limits are restored, with regulating CEAs not within the applicable insertion and withdraw sequence limits. The CTS 3.1.3.1 ACTION d.2 operational limit "for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year," is inconsistent with the requirements of CTS 3.0.2 (ITS LCO 3.0.2). CTS 3.0.2 states, in part, "If the LCO is met or is no longer applicable prior to expiration of the specified time interval(s), completion of the ACTIONS is not required, unless otherwise stated." Once the CEA alignment limits have been restored, LCO 3.1.3.1 (ITS LCO 3.1.4) is met and, therefore, the remaining action limiting operation to, "7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year," is no longer required since there is no note requiring completion of this action following restoration of the LCO. This change is designated as administrative and has no technical impact because the operational limit "for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year," is an action that is never applicable since the CTS (and ITS) ACTIONS require the CEA Alignment LCO to be met prior to the applicable action requirement becoming limiting.

A08 CTS 4.1.3.1.4.a states that the CEA Block Circuit be demonstrated OPERABLE prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 92 days. The associated footnote (*) states that the licensee shall be excepted from compliance during the initial startup test program for an entry into MODE 2 from MODE 3 made in association with a measurement of power defect.

The purpose of the CTS statement is to provide an exception to performance of the CEA Block Circuit Surveillance at less than a 92 day Frequency. CTS 4.0.4 (ITS SR 3.0.4) requires LCO's Surveillances have been met within their specified frequency prior to entry into a MODE or other specified condition in the Applicability of an LCO, except as provided by CTS 4.0.3 (ITS SR 3.0.3). Therefore, the requirement to perform the Surveillance prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 92 days, is duplicative and unnecessary. CTS 4.1.3.1.4.a (ITS SR 3.1.4.2) will continue to be performed prior to entry into

MODE 2 from MODE 3 and within the Frequency specified in the Surveillance Frequency Control Program.

The purpose of the CTS footnote (*) is to provide a special test exception (STE). CTS 3.10.2, Group Height, Insertion and Power Distribution Limits (STE) provide provision for suspension of certain Specifications, including CTS 3.1.3.1, for startup testing. In addition, ITS LCO 3.0.7 provides guidance on the use of special test exceptions. It is an ITS convention to not include duplicative requirements from other Specifications. ITS 3.1.8 retains the CTS 3.10.2 requirements, therefore, the CTS footnote (*) is unnecessary and deleted.

This changes the CTS by deleting the statement to perform the Surveillance prior to entry into MODE 2 from MODE 3, except performance of the CEA Block Circuit Surveillance at less than a 92 day Frequency is not required, and deleting the footnote (*) that provides a special test exception. This change is designated as administrative as it removes duplicative requirements to Section 3.0 usage requirements and other Specifications with no technical change to the CTS.

A09 Unit 1 CTS 3.1.3.4 Action a. states “With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.” Unit 2 CTS 3.1.3.4 Action a.2 states “If in MODE 3, 4, or 5, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.” In both cases, the “above limit’ refers to the CEA drop time specified in CTS 3.1.3.4. CTS 3.0.4 (ITS LCO 3.0.4) states that when an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made under certain conditions. The purpose of CTS 3.1.3.4 is to ensure the CEA drop times are determined within limits prior to entering MODES 1 and 2. Since a plant shutdown (i.e., be in MODE 3 within 6 hours) is required when one or more CEAs is not within the CEA drop time, CTS 3.1.3.4 Actions (ITS 3.1.4 ACTIONS) do not meet any of the LCO 3.0.4 conditions, and therefore, the LCO must be met prior to entry into the MODE of applicability. These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A10 CTS 3.1.3.4 ACTION b provides an allowance for drop times to be determined at less than full reactor coolant flow provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of the CEA Drop time. ITS 3.1.4 does not contain a similar allowance. This changes the CTS by not allowing continued operation at reduced power when the CEA drop times are determined with only 3 reactor coolant pumps operating. This change is designated as administrative change and is acceptable because they do not result in technical changes to the CTS.

The purpose of CTS 3.1.3.4 is to ensure the CEAs insert within the CEA drop criteria. This change is acceptable because ITS SR 3.1.4.5 requires verification of the CEA drop times be performed with all of the Reactor coolant pumps operating. In accordance with CTS 3.4.1.1 and ITS 3.4.1, Unit 1 and Unit 2 only operate in MODES 1 and 2 with both reactor coolant loops and both reactor coolant pumps in each loop in operation. Therefore, ITS 3.1.4 will not allow the CEA drop times to be determined with only 3 reactor coolant pumps operating.

This change is designated as administrative change and is acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 4.1.3.1.1 states that with the CEA Deviation Circuit and/or the CEA Block Circuit inoperable, verification of the individual CEA positions is required once per 4 hours. However, there is no explicit requirement for the CEA Deviation Circuit to be OPERABLE and no periodic Surveillance is specified to verify the CEA Deviation Circuit is OPERABLE. ITS LCO 3.1.4 requires, in part, that the CEA deviation circuit be OPERABLE and ITS SR 3.1.4.3 requires verification that the deviation circuit is OPERABLE with a Frequency in accordance with the Surveillance Frequency Control Program (SFCP). This changes the CTS by explicitly requiring the deviation circuit be OPERABLE and adding a Surveillance Requirement to verify the deviation circuit is OPERABLE with a Frequency in accordance with the SFCP.

The purpose of the CEA Deviation Circuit is to ensure improper CEA alignments are identified before unacceptable flux distributions occur. CTS 4.1.3.1.1 implies that the CEA deviation circuit is required to be OPERABLE. This change is acceptable because OPERABILITY requirements, implicit in the CTS, and a periodic Surveillance are added to the ITS. FPL adopted an SFCP in Amendments 223 and 173 for PSL Units 1 and 2, respectively (ADAMS Accession No. ML15127A066). Therefore, the Frequency of ITS SR 3.4.1.3 will be in accordance with the SFCP with an initial periodic frequency of 92 days consistent with the Frequency specified in ISTS SR 3.1.4.3. This Frequency considers other information continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA motion inhibit.

This change has been designated as more restrictive because it explicitly adds an LCO operability requirement and Surveillance Requirement to the CTS.

M02 CTS 3.1.3.1 Unit 1 Action f, and Unit 2 Action e require, in part, THERMAL POWER to be reduced ≤ 70 RTP prior to performing Actions f.1 or f.2 (Unit 1) and Action e.1 or e.2 (Unit 2). After the power reduction, Action f.1 (Unit 1) and Action e.1 (Unit 2) require restoring the CEA to within the required alignment limits. The time to reduce THERMAL POWER and the time to restore CEA alignment limit in Action f.1 (Unit 1) and Action e.1 (Unit 2) are not defined in CTS. ITS 3.1.4, Required Action C.1 requires 1 hour to reduce THERMAL POWER to ≤ 70 RTP and Required Action C.2 requires a total of 2 hours (combining the 1 hour to reduce power and 1 hour to restore CEA alignment) to restore the CEA to within the required alignment limits. This changes CTS by adding a specific Completion Time to reduce THERMAL POWER and restore CEA alignment to within limits.

The purpose of CTS 3.1.3.1 actions, in part, is to ensure, when a major CEA alignment deviation occurs, that the CEA alignment be restored in a reasonable period of time to ensure core power distributions are maintained and thermal limits are not exceeded. Xenon redistribution in the core starts to occur as soon

as a CEA becomes misaligned. If the alignment cannot be restored within one hour, a power reduction is required to minimize the effects of the xenon redistribution, while allowing additional time to restore the CEA alignment consistent with the time provided in CTS 3.1.3.1 Unit 1 Action f.2 and Unit 2 Action e.2.

ITS 3.1.4, ACTION C requires THERMAL POWER to be reduced ≤ 70% RTP within 1 hour and the CEA to be restored to within the alignment limits within 2 hours. The Completion Time of Required Action C.1 provides the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System. The Completion Time of C.2 is sufficient to take appropriate corrective action to realign the CEA following the power reduction while recognizing the importance of minimizing the effects of xenon redistribution.

This change has been designated as more restrictive because it adds to the CTS actions an explicit time to complete the THERMAL POWER reduction and time to restore the CEA to within the required limit following the power reduction.

M03 CTS 4.1.3.1.1 states that with the CEA Deviation Circuit and/or the CEA Block Circuit inoperable, verification of the individual CEA positions is required once per 4 hours. ITS 3.1.4 ACTION D and ACTION E require, with the CEA block circuit or the CEA deviation circuit inoperable, ITS SR 3.1.4.1 (verification of the individual CEA positions) must be performed within 1 hour and once per 4 hours thereafter. This changes the CTS by requiring the initial verification of the individual CEA positions be performed within 1 hour.

The purpose of CTS 4.1.3.1.1 CEA Block Circuit is to permit CEA motion within the requirements of LCO 3.1.6, "Regulating Control Element Assembly Insertion Limits," and prevents regulating CEAs from being misaligned from other CEAs in the group. The purpose of CTS 4.1.3.1.1 CEA Deviation Circuit is to ensure improper CEA alignments are identified before unacceptable flux distributions occur. Performing ITS SR 3.1.4.1 within 1 hour and every 4 hours thereafter, is considered acceptable since it takes into account other information continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and the protection provided by the CEA block and deviation circuit is not required. This change has been designated as more restrictive because it reduces the amount of time to perform the initial verification of CEA positions.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.1.3.1 Action a. requires that with one or more full-length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the

SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours. This changes the CTS by relocating details for CEA operability. The details for CEA inoperability due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable is relocated to the ITS Bases. The removal of these details that are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for CEA operability. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.1.3.1 Unit 1 Action e and Unit 2 Action d state that a misaligned CEA be positioned within the LCO alignment limit of other CEAs in its group in accordance with the time constraints shown in COLR Figure 3.1-1a. ITS 3.1.4 Required Action B.1 requires that the CEA alignment be restored "in accordance with the COLR" when one CEA is misaligned more than the maximum allowable alignment deviation. This changes the CTS by relocating details of the specific location in the COLR where the Completion Time value is provided. The details of the COLR figure number where a time restraints are provided based on the measured F_r^T value prior to the misalignment are relocated to the ITS Bases.

The purpose of CTS 3.1.3.1 Unit 1 Actions e and f and Unit 2 Actions d and e is to ensure CEA alignment is restored in a reasonable period of time to ensure core power distributions are maintained and thermal limits are not exceeded. Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned.

The removal of these details that are related to a specific document reference, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to restore a misaligned CEA to within the required alignment limit with a Completion Time in accordance with the COLR. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications

LA03 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 4.1.3.1.3 requires the CEA Block Circuit to be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by a functional test which verifies that the circuit prevents any CEA from being | R1

misaligned from all other CEAs in its group by more than the required limit (indicated position).

CTS 4.1.3.1.4 requires the CEA Block Circuit to be demonstrated OPERABLE by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit prevents the regulating CEAs from being inserted beyond the Power Dependent Insertion Limit of COLR Figure 3.1-2.

ITS SR 3.1.4.2 does not retain this detail. This changes the CTS by relocating the details that a functional test that verifies the CEA Block Circuit prevents any CEA from being misaligned from all other CEAs in its group by more than the LCO limit (indicated position), and verifies the CEA Block Circuit circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 (ITS 3.1.6) and the CEA Block Circuit prevents the regulating CEAs from being inserted beyond the Power Dependent Insertion Limit of COLR Figure 3.1-2 to the Bases.

The removal of these details, that are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the Surveillance requirement that the CEA Block Ciruit be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA04 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.1.3.4 requires the individual (shutdown and regulating) CEA drop time from the fully withdrawn position shall be less than or equal to 3.1 seconds (Unit 1) and less than or equal to 3.25 seconds (Unit 2) from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches the 90% insertion position with T_{avg} greater than or equal to 515°F and all reactor coolant pumps operating. This changes the CTS by relocating details for the CEA drop time surveillance. The CEA drop time of less than or equal to 3.1 seconds is provided in Unit 1 ITS SR 3.1.4.5, and the CEA drop time of less than or equal to 3.25 seconds is provided in Unit 2 ITS SR 3.1.4.5, rather than the ITS 3.1.4 Bases description for SR 3.1.4.5. Because Units 1 and 2 only operate with four reactor coolant pumps, this detail is deleted and not included in the CEA drop time details provided in the ITS 3.1.4 Bases description for SR 3.1.4.5.

The removal of these details that are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for performing CEA drop time testing from the fully withdrawn position. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the

Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.1.3.1 ACTION a. states that with a CEA inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, satisfy the SHUTDOWN MARGIN (SDM) requirements of Specification 3.1.1.1 within 1 hour. Unit 1 CTS 3.1.3.1 ACTION d.2., d.2.b., f.2., f.2.b., state that with a misaligned CEA not restored to alignment requirements within 1 hour, declare the CEA inoperable and satisfy SDM requirements of Specification 3.1.1.1. Additionally, the SDM requirement of Specification 3.1.1.1 and 3.1.1.2 is determined at least once per 12 hours thereafter while the CEA(s) are inoperable in accordance with CTS 4.1.1.1.1.a and CTS 4.1.1.2.a, respectively.

Unit 2 CTS 3.1.3.1 ACTION a. states that with a CEA inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, satisfy the SDM requirements of Specification 3.1.1.1 within 1 hour. Unit 2 CTS 3.1.3.1 ACTION e.2., e.2.b., f.2., f.2.b., state that with a misaligned CEA not restored to alignment requirements within 1 hour, declare the CEA inoperable and satisfy SDM requirements of Specification 3.1.1.1. Additionally, the SDM requirement of Specification 3.1.1.1 and 3.1.1.2 is determined at least once per 12 hours thereafter while the CEA(s) are inoperable in accordance with CTS 4.1.1.1.1.a and CTS 4.1.1.2.a, respectively.

CTS 3.1.1.1, SDM requirements are applicable in MODES 1, 2, 3, and 4. ITS 3.1.4 does not require a SDM verification when one or more CEAs are inoperable. This changes the CTS Actions by not requiring SDM verification within 1 hour and once per 12 hours thereafter with one or more inoperable CEAs.

The purpose of CTS 4.1.1.1.1.a and CTS 4.1.1.2.a is to provide the appropriate compensatory measures to determine SDM when CEAs are inoperable during operations in MODES 1, 2, 3, 4, and 5. The purpose of the ITS 3.1.4 ACTIONS are to provide the appropriate compensatory actions for inoperable CEAs in MODES 1 and 2. ITS 3.1.5 and 3.1.6 provide requirements for shutdown and regulating CEA insertion limits in MODES 1 and 2. These insertion limits ensure SDM is maintained while the unit is at power. The purpose of ITS SR 3.1.1.1 is to periodically verify SDM regardless of the status of the CEAs in MODES 3, 4, and 5. ITS SR 3.1.1.1 requires SDM verification in accordance with the Surveillance Frequency Control Program. When the plant is operating in MODES 1 and 2, with one or more CEAs inoperable, the unit cannot continue operation and must be placed in MODE 3 within 6 hours. After reaching MODE 3, ITS 3.1.4 no longer applies therefore it is inappropriate to specify additional actions after the unit is outside the Applicability of the Specification. Nevertheless, SDM is periodically verified in accordance with ITS SR 3.1.1.1 in

MODES 3, 4, and 5 in accordance with the frequency specified in the Surveillance Frequency Control Program.

This SDM verification must also compensate for the reactivity worth of the CEA as specified in the definition of SDM. This change is acceptable since SDM will still be required to be monitored at a frequency in accordance with the Surveillance Frequency Control Program, and based on the definition of SDM, the reactivity worth of CEA not capable of being fully inserted must be accounted for in the determination of SDM. Since the proposed actions continue to require a plant shutdown to at least MODE 3 and a verification of SDM must account for the reactivity worth of an inoperable CEA, sufficient remedial measures continue to be provided to allow safe operation pursuant the requirements of 10 CFR 50.36(c)(2). SDM continues to be monitored in a manner and at a Frequency necessary to give confidence that the assumptions in the safety analyses are protected. The frequency currently established for the SDM margin verification in accordance with the Surveillance Frequency Control Program is 24 hours. This change is designated as less restrictive because an Action to perform a Surveillance which is required in the CTS will not be explicitly required in the ITS.

L02 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.1.3.1 Action b and b.2 states, in part, with the CEA Block Circuit inoperable, within 6 hours, with the group overlap and/or sequencing interlocks inoperable, maintain CEAs groups 3, 4, 5 and 6 fully withdrawn and withdraw the CEAs in group 7 to less than 5% insertion and place and maintain the CEA drive system mode switch in either the "Manual" or "Off" position. Unit 2 CTS 3.1.3.1 Action b and b.2 states, in part, with the CEA Block Circuit inoperable, within 6 hours, with the group overlap and/or sequencing interlocks inoperable, maintain CEA groups 1, 2, 3 and 4 fully withdrawn and the CEAs in group 5 to less than 15% insertion and place and maintain the CEA drive system in either the "Manual" or "Off" position. ITS 3.1.4 Required Action D.2.2 retains the CTS requirement, however, the ITS adds a Note that states that Required Action D.2.2 shall not be performed when in conflict with either Required Action C.1, C.2, or E.1. The Note is less restrictive by not requiring performance of Required Action D.2.2 when in conflict with reduction in THERMAL POWER to ≤ 70% RTP per Required Action C.1, restoring CEA alignment to within limits per Required Action C.2, or when performing the deviation circuit Surveillance (SR 3.1.4.1) per Required Action E.1. This changes the CTS by relaxing the requirements for an Action.

The purpose of Unit 1 CTS 3.1.3.1 Actions is to ensure acceptable power distributions are maintained and core thermal limits are not exceeded. This change is acceptable because the actions in the Note are consistent with safe operation under the specific Condition, considering the actions of lowering THERMAL POWER, restoring CEA alignment to within limits, or verifying the deviation circuit is OPERABLE by performance of the Surveillance, are sufficient remedial actions to ensure thermal limits are not exceeded and, therefore, should not be prevented by Required Action D.2.2. This change is designated as less restrictive because the requirements for an Action are relaxed.

L03 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.3.4.b requires the CEA drop time of CEAs shall be demonstrated through measurement prior to reactor criticality for specifically affected individual CEAs following any

maintenance on or modification to the CEA control system which could affect the drop time of those specific CEAs. ITS 3.1.4 does not contain this testing requirement. This changes the CTS by not explicitly requiring post-maintenance testing on CEAs.

The purpose of CTS 4.1.3.4.b is to verify OPERABILITY of the CEAs following maintenance that could alter their operation. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. Any time the OPERABILITY of a system or component has been affected by repair, maintenance, modification, or replacement of a component, post-maintenance testing is required to demonstrate the OPERABILITY of the system or component. This is described in the Bases for ITS SR 3.0.1 and required under ITS SR 3.0.1. The OPERABILITY requirements for the CEA control system are described in the Bases for ITS 3.1.4. In addition, the requirements of 10 CFR 50, Appendix B, Section XI (Test Control) provide adequate controls for test programs to ensure that testing incorporates applicable acceptance criteria. Compliance with 10 CFR 50, Appendix B, is required under the unit operating license. As a result, post-maintenance testing will continue to be performed and an explicit requirement in the Technical Specifications is not necessary. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L04 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.3.4 requires drop testing of CEAs to be demonstrated through measurement prior to reactor criticality following each removal of the reactor vessel head and in accordance with the Surveillance Frequency Control Program, which is at least once per 18 months. ITS SR 3.1.4.5 requires the test to be performed prior to criticality after each removal of the reactor head. This changes the CTS by deleting the requirement to perform this test at least once per 18 months. | R1

The purpose of CTS 4.1.3.4 is to ensure the CEA insert within the CEA drop criteria. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its safety function. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence the equipment can perform its assumed safety function. The requirements in the CTS to perform the test following each removal of the reactor vessel head and at least once per 18 months normally coincide with one another. The head is removed once per 18 months unless there is a need to remove the head prior to the end of the cycle. This change is designated as less restrictive because a Surveillance that was required in the CTS will not be performed in the ITS.

L05 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* When a CEA is declared inoperable when misaligned by more than 15 inches and beyond the time constraints specified in the COLR figure, CTS 3.1.3.1 e.2 footnote (*) requires that if the pre-misalignment AXIAL SHAPE INDEX (ASI) was more negative than –0.15, power must be reduced to < 70% of RATED THERMAL POWER or 70% of the THERMAL POWER level prior to the misalignment,

whichever is less, prior to completing ACTION e.2.a) and e.2.b). ITS 3.1.4 does not include this additional power reduction requirement based on ASI. This changes the CTS by eliminating the requirement to reduce power to 70% of the THERMAL POWER level prior to the misalignment.

The purpose of the CTS 3.1.3.1 e.2 footnote (*) is to ensure adequate margin to thermal limits are maintained during recovery from a major CEA alignment deviation with a negative ASI value. CTS 3.2.5 (ITS 3.2.4) provides the appropriate remedial actions for conditions when ASI is outside limits specified in the COLR to ensure thermal limits are not exceeded. This change is acceptable because the CTS 3.2.5 action requirements, retained in ITS 3.2.4, provide the appropriate remedial actions for conditions when ASI is outside limits specified in the COLR to ensure thermal limits are not exceeded. The time to restore ASI to within limits specified in ITS 3.2.4 ACTION A is considered sufficiently short that the xenon distribution in the core cannot change significantly and the action to reduce THERMAL POWER in ITS 3.2.4 ACTION B provides reasonable assurance that the core is operating farther from thermal limits and places the core in a conservative condition. This change is designated as less restrictive because the requirements for an Action are relaxed.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

3.1.4 Control Element Assembly (CEA) Alignment ~~(Analog)~~

3.1.3.1 LCO 3.1.4 All CEAs shall be OPERABLE.

AND

All CEAs shall be aligned to within ~~[7]~~ 7.5 inches (indicated position) of their respective group, and [the CEA ~~motion inhibit~~ block circuit and the CEA deviation circuit shall be OPERABLE].

3.1.3.1 DOC M01

Applicability APPLICABILITY: MODES 1 and 2.

ACTIONS

Insert 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A.~~ ~~One or more CEAs misaligned from its group by > [7 inches] and ≤ [15 inches].~~ ~~OR~~ ~~One CEA misaligned from its group by > [15 inches].~~	~~A~~.1 Reduce THERMAL POWER to ≤ 70% RTP. AND ~~A~~.2 Restore CEA ~~Alignment~~ to within alignment limit.	1 hour 2 hours
~~B.~~ CEA ~~motion inhibit~~ block circuit inoperable.	~~B~~.1 Perform SR 3.1.4.1. AND ~~B~~.2.1 Restore CEA ~~motion inhibit~~ block circuit to OPERABLE status. OR	1 hour AND Every 4 hours thereafter 6 hours

3.1.3.1 Action f DOC M02 DOC L01

Insert 2

3.1.3.1 Action b

4.1.3.1.1 DOC M03

3.1.3.1 Action b

CTS

3

INSERT 1

3.1.3.1 Action d	A. One or more CEAs misaligned from its group by > 7.5 inches and < 15 inches.	A.1	Restore CEAs to within alignment limit.	2 hours
3.1.3.1 Action e	B. One CEA misaligned from its group by ≥ 15 inches.	B.1	Restore CEA alignment to < 15 inches.	In accordance with the COLR

3

INSERT 2

3.1.3.1 Action f	Required Action and associated Completion Time of Condition B not met.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~B~~.2.2 ----------NOTE---------- Required Action ~~B~~.2.2 shall not be performed when in conflict with either Required Action ~~A.1, A.2, or C.1~~. ---------- ~~P~~lace and maintain the CEA drive switch in either the "off" or "manual" position~~, [and f~~ully withdraw all CEAs in groups 3 and ~~4~~ and withdraw all CEAs in group ~~5~~ to < 5% insertion~~]~~.	6 hours
~~C~~. CEA deviation circuit inoperable.	~~C~~.1 Perform SR 3.1.4.1.	1 hour AND Every 4 hours thereafter
~~D~~. Required Action and associated Completion Time of Conditions A, ~~B, or~~ C not met. OR One or more CEAs inoperable. OR Two or more CEAs misaligned by ~~>~~ ~~[~~15 inches~~]~~.	~~D~~.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.1.3.1.1	SR 3.1.4.1	Verify the indicated position of each CEA to be within ~~[~~7 inches~~]~~ of all other CEAs in its group. (7.5)	Within 1 hour following any CEA movement of > ~~[~~7 inches~~]~~ (7.5) AND ~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.1.3.1.3 4.1.3.1.4	SR 3.1.4.2	Verify the CEA ~~motion inhibit~~ block circuit is OPERABLE.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	4 2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.1.3.1.1	SR 3.1.4.3	Verify the CEA deviation circuit is OPERABLE.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.1.3.1.2	SR 3.1.4.4	Verify CEA freedom of movement (trippability) by moving each individual CEA that is not fully inserted into the reactor core ~~[~~5 inches~~]~~ in either direction. (7.5)	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	2
	~~SR 3.1.4.5~~	~~Perform a CHANNEL FUNCTIONAL TEST of the reed switch position transmitter channel.~~	~~[18 months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~	5
4.1.3.4 DOC L03 DOC L04	SR 3.1.4.~~6~~ (5)	Verify each CEA drop time is ≤ ~~[~~3.1~~]~~ ~~seconds~~.	Prior to reactor criticality, after each removal of the reactor head and installation	4 2 5

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

3.1.4 Control Element Assembly (CEA) Alignment ~~(Analog)~~

3.1.3.1 LCO 3.1.4 All CEAs shall be OPERABLE.

AND

All CEAs shall be aligned to within ~~[~~7~~]~~ inches (indicated position) of their respective group, and ~~[~~the CEA ~~motion inhibit~~ and the CEA deviation circuit shall be OPERABLE~~]~~.

3.1.3.1 DOC M01

block circuit

Applicability APPLICABILITY: MODES 1 and 2.

Insert 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A.~~ ~~One or more CEAs misaligned from its group by > [7 inches] and ≤ [15 inches].~~ ~~OR~~ ~~One CEA misaligned from its group by > [15 inches].~~	~~A~~.1 Reduce THERMAL POWER to ≤ 70% RTP. AND ~~A~~.2 Restore CEA ~~Alignment~~.	1 hour 2 hours
~~B~~. CEA ~~motion inhibit~~ inoperable.	~~B~~.1 Perform SR 3.1.4.1.	1 hour AND Every 4 hours thereafter
	AND ~~B~~.2.1 Restore CEA ~~motion inhibit~~ to OPERABLE status. OR	6 hours

Insert 2

3.1.3.1 Action e, f
DOC M02
DOC L01
DOC L05

3.1.3.1 Action b

4.1.3.1.1 DOC M03

3.1.3.1 Action b.1

block circuit

to within alignment limit

CTS

3

INSERT 1

3.1.3.1 Action f	A. One or more CEAs misaligned from its group by > 7 inches and ≤ 15 inches.	A.1	Restore CEAs to within alignment limit.	2 hours
3.1.3.1 Action d	B. One CEA misaligned from its group by > 15 inches.	B.1	Restore CEA alignment to ≤ 15 inches.	In accordance with the COLR

3

INSERT 2

3.1.3.1 Action e

Required Action and associated Completion Time of Condition B not met.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~B~~.2.2 ----------NOTE---------- Required Action ~~B~~.2.2 shall not be performed when in conflict with either Required Action ~~A.1, A.2, or C.1~~. ---------- ~~P~~lace and maintain the CEA drive switch in either the "off" or "manual" position~~, [and f~~ully withdraw all CEAs in groups 3 and 4 and withdraw all CEAs in group 5 to < ~~5~~% insertion~~]~~.	6 hours
~~C~~. CEA deviation circuit inoperable.	~~C~~.1 Perform SR 3.1.4.1.	1 hour AND Every 4 hours thereafter
~~D~~. Required Action and associated Completion Time of Conditions A~~, B, or~~ C not met. OR One or more CEAs inoperable. OR Two or more CEAs misaligned by > ~~[~~15 inches~~]~~.	~~D~~.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.1.3.1.1	SR 3.1.4.1	Verify the indicated position of each CEA to be within ~~[~~7 inches~~]~~ of all other CEAs in its group.	Within 1 hour following any CEA movement of > ~~[~~7 inches~~]~~ AND ~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ (2)
4.1.3.1.3 4.1.3.1.4	SR 3.1.4.2	Verify the CEA ~~motion inhibit~~ block circuit is OPERABLE.	~~[92 days~~ (4) ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ (2)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.1.3.1.1	SR 3.1.4.3	Verify the CEA deviation circuit is OPERABLE.	~~[~~ 92 days ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2
4.1.3.1.2	SR 3.1.4.4	Verify CEA freedom of movement (trippability) by moving each individual CEA that is not fully inserted into the reactor core ~~[~~5 inches~~]~~ in either direction. 7	~~[~~ 92 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 2
	~~SR 3.1.4.5~~	~~Perform a CHANNEL FUNCTIONAL TEST of the reed switch position transmitter channel.~~	~~[18 months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~ 5
4.1.3.4 DOC L03 DOC L04	SR 3.1.4.~~6~~ 5	Verify each CEA drop time is ≤ ~~[3.1]~~ seconds. 3.25	Prior to reactor criticality, after each removal of the reactor head and installation 4 2 5

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.4, CONTROL ELEMENT ASSEMBLY (CEA) ALIGNMENT

1. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering (CE) vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ACTIONS A and B are added in the PSL Unit 1 and Unit 2 ITS. ITS ACTION A provides an action to restore the CEA alignment to within the limit in the condition of a minor CEA alignment deviation. For a major CEA alignment deviation, ITS ACTION B first allows time to restore the CEA alignment to within limits prior to the power reduction with a Completion Time of "in accordance with the COLR." These changes are consistent with the CTS Actions. The CTS Action requirement for a minor CEA alignment deviation does not require a THERMAL POWER reduction. For a major CEA alignment deviation, the CTS Actions allow time to restore CEA alignment prior to requiring a THERMAL POWER reduction, provided the time constraints in the specified COLR figure are not exceeded. This allowance was approved in License Amendments 150 and 92 for Unit 1 and Unit 2, respectively (ADAMS Accession No. ML17229A288 (U1) and ML013600531 (U2)) and is considered acceptable because the Completion Time in accordance with the COLR is dependent upon the F_r^T prior to the major CEA alignment deviation. A worst case analysis has previously shown that a DNB fuel design limit violation may occur during this condition but is eliminated by limiting the time operation is permitted at full power before a power reduction is required. Changes to the time constraint specified in the COLR are controlled by an NRC-approved methodology. This change to the ISTS is also similar to the ITS 3.1.4 ACTIONS of another analog type CE plant; Calvert Cliff Nuclear Power Plant, Units 1 and 2 (ADAMS Accession No. ML052720276). ISTS 3.1.4, Condition A (ITS 3.1.4., Condition C) is revised to state, "Required Action and associated Completion Time of Condition B not met. ISTS Required Action A.2 (ITS Required Action C.2) is revised to be consistent with the wording in Required Action A.1 of new ITS ACTION A and is considered editorial. The Completion Time for ISTS Required Actions A.1 and A.2 (ITS Required Actions C.1 and C.2) remain unchanged and are consistent with the Calvert Cliffs ITS 3.1.4 ACTION C Completion Times. Subsequent Conditions, Required Actions, and references to Required Actions are renumbered to reflect this change.

4. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

5. ISTS SR 3.1.4.5 is deleted and subsequent Surveillance is renumbered. PSL developed ITS 3.1.7, Control Element Assembly (CEA) Position Indication, and requires performance of ITS SR 3.1.7.1 to determine operability of the reed switch position indicating channels and the pulse counting position indicating channels in accordance with the Surveillance Frequency Control Program.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

1

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

1

B 3.1.4 Control Element Assembly (CEA) Alignment ~~(Analog)~~

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and regulating Control Element Assemblies (CEAs) is an initial assumption in all safety analyses that assume CEA insertion upon reactor trip. Maximum CEA misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10 and GDC 26 (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a CEA to become inoperable or to become misaligned from its group. CEA inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CEA worth for reactor shutdown. Therefore, CEA alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on CEA alignment and OPERABILITY have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CEAs are moved by their control element drive mechanisms (CEDMs)~~. Each CEDM moves its CEA~~ one step (approximately ¾ inch) at a time~~, but at varying rates (steps per minute) depending on the signal output from the Control Element Drive Mechanism Control System (CEDMCS)~~.

2

The CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating CEAs also provide reactivity (power level) control during normal operation and transients. ~~Their movement may be automatically controlled by the Reactor Regulating System.~~

2

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems, which are the ~~Plant Computer CEA~~ Position Indication System and the Reed Switch Position Indication System.

3

Pulse Counting

3

1

BASES

BACKGROUND (continued)

Pulse Counting

The ~~Plant Computer CEA~~ Position Indication System counts the commands sent to the CEA gripper coils from the CEDM Control System that moves the CEAs. There is a one step counter for each ~~group of~~ CEA~~s~~. Individual CEAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. ~~Plant Computer CEA~~ Position Indication System is considered highly precise (± 1 step or ± ¾ inch). ~~If a CEA does not move one step for each command signal, the step counter will still count the command and incorrectly reflect the position of the CEA.~~

3

An individual CEA may be moved in the Manual Individual Mode. The Pulse Counting

A step is counted after the Step Complete signal is received.

3 4

4

The Reed Switch Position Indication System provides a highly accurate indication of actual CEA position, but at a lower precision than the step counters. This system is based on ~~inductive analog~~ signals from a series of reed switch~~es~~ spaced ~~along a tube with a center to center distance of~~ 1.5 inch~~es~~, which is two steps. ~~To increase the reliability of the system, there are redundant reed switches at each position.~~

magnetically actuated

position transmitters

at

intervals

4

3

APPLICABLE SAFETY ANALYSES

CEA misalignment accidents are analyzed in the safety analysis (Ref. 3). The accident analysis defines CEA misoperation as an~~y~~ event~~, with the exception of sequential group withdraws,~~ which could result from a single malfunction in the reactivity control systems. ~~For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDMCS, or by operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper. Inadvertent withdrawal of a single CEA may be caused by the opening of the electrical circuit of the CEDM holding coil for a full length or part length CEA. A dropped CEA could be caused by an electrical failure in the CEA coil power programmers~~.

5

The acceptance criteria for addressing CEA inoperability or misalignment are that:

a. There shall be no violations of either:

   1. Specified acceptable fuel design limits or

   2. Reactor Coolant System (RCS) pressure boundary integrity and

b. The core must remain subcritical after accident transients.

3

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~Three types of~~ misalignment are distinguished in the safety analysis (Ref. 1). ~~During movement of a group, one CEA may stop moving while the other CEAs in the group continue. This condition may cause excessive power peaking. The second type of~~ misalignment occurs if one CEA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining CEAs to meet the SDM requirement with the maximum worth CEA stuck fully withdrawn. If a CEA is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck CEAs into account. ~~The third type of~~ misalignment occurs when one CEA drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power, due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

~~Two types of~~ analyses are performed in regard to static CEA misalignment (Ref. ~~4~~). With CEA banks at their insertion limits, one type of analysis considers the case when any one CEA is inserted ~~[~~ ~~]~~ inches into the core. The second type of analysis considers the case of a single CEA withdrawn ~~[~~ ~~]~~ inches from a bank inserted into its insertion limit. Satisfying limits on departure from nucleate boiling ratio (DNBR) in both of these cases bounds the situation when a CEA is misaligned from its group by ~~[7~~ inches~~]~~.

Another type of misalignment occurs if one CEA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth CEA also fully withdrawn (Ref. ~~5~~).

Since the CEA drop incidents result in the most rapid approach to specified acceptable fuel design limits (SAFDLs) caused by a CEA misoperation, the accident analysis analyzed a single full length CEA drop. The most rapid approach to the DNBR SAFDL may be caused by a single full length CEA drop or a CEA subgroup drop, depending upon initial conditions.

~~All of the above CEA misoperations will result in an automatic reactor trip.~~ In the case of the full length CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters.

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The results of the CEA misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or RCS pressure occur.

CEA alignment limits and OPERABILITY requirements satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on shutdown and regulating CEA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on CEA OPERABILITY ensure that upon reactor trip, the CEAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The CEA OPERABILITY requirements (i.e., trippability) are separate from alignment requirements which ensure that the CEA banks maintain the correct power distribution and CEA alignment. The CEA OPERABILITY requirement is satisfied provided the CEA will fully insert in the required CEA drop time assumed in the safety analysis. CEA control malfunctions that result in the inability to move a CEA (e.g., CEA lift rod failures), but do not impact trippability, do not result in CEA inoperability.

7.5

The requirement is to maintain the CEA alignment to within ~~[7~~ inches~~]~~ between any CEA and its group. The minimum misalignment assumed in safety analysis is ~~[~~15 inches~~]~~, and in some cases, a total misalignment from fully withdrawn to fully inserted is assumed.

6

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMS, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on CEA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of CEAs have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the CEAs are bottomed, and the reactor is shut down and not producing fission power. In the shutdown Modes, the OPERABILITY of the shutdown and regulating CEAs has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

3

BASES

ACTIONS

A.1 ~~and A.2~~

If one or more CEAs (regulating or shutdown) are misaligned by > ~~[~~7 inches~~]~~ and ≤ [15 inches~~]~~, ~~or one CEA is misaligned by > [15 inches], continued operation in MODES 1 and 2 may continue, provided, within 1 hour, the power is reduced to ≤ 70% RTP, and within~~ 2 hours ~~CEA alignment is restored~~. Regulating CEA alignment can be restored by either aligning the misaligned CEA(s) to within ~~[~~7 inches~~]~~ of its group or aligning the misaligned CEA's group to within ~~[~~7 inches~~]~~ of the misaligned CEA. Shutdown CEA alignment can be restored by aligning the misaligned CEA(s) to within ~~[~~7 inches~~]~~ of its group.

~~Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Reducing THERMAL POWER in accordance with Figure 3.1.4-1 (in the associated LCO) ensures acceptable power distributions are maintained (Ref. 6).~~ For ~~small misalignments~~ (< ~~[~~15 inches~~]~~) of the CEAs, there is:

a. A small effect on the time dependent long term power distributions relative to those used in generating LCOs and limiting safety system settings (LSSS) setpoints,

b. A negligible effect on the available SDM, and

c. A small effect on the ejected CEA worth used in the accident analysis.

~~With a large CEA misalignment (≥ [15 inches]), however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints. The effect on the available SDM and the ejected CEA worth used in the accident analysis remain small. Therefore, this condition is limited to a single CEA misalignment, while still allowing 2 hours for recovery.~~

~~In both cases, a~~ 2 hour~~s~~ ~~time period~~ is sufficient to:

a. Identify cause of a misaligned CEA, ~~and~~

b. Take appropriate corrective action to realign the CEAs~~, and~~

~~c. Minimize the effects of xenon redistribution~~.

Bases Insert 1

8

BASES INSERT 1

B.1

If one CEA is misaligned from its group by ≥ 15 inches, CEA alignment must be restored to < 15 inches in 1 hour. Regulating CEA alignment can be restored by either aligning the misaligned CEA to < 15 inches of its group or aligning the misaligned CEA's group to < 15 inches of the misaligned CEA. Shutdown CEA alignment can be restored by aligning the misaligned CEA to < 15 inches of its group.

Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. With a major CEA alignment deviation (≥ 15 inches), misalignment could cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints. The effect on the available SDM and the ejected CEA worth used in the accident analysis remain small. Therefore, this condition is limited to a single major CEA alignment deviation and the time constraints provided in the applicable COLR figure for recovery. A higher total integrated peaking factor (F_r^T) prior to the misalignment could result in a shorter time constraint, further limiting the Completion Time before requiring a power reduction.

The Completion Time in accordance with the COLR is dependent upon the F_r^T prior to the major CEA alignment deviation. A worst case analysis has shown that a DNBR SAFDL violation may occur during this condition. This potential DNBR SAFDL violation is eliminated by limiting the time operation is permitted at full power before a power reduction is required.

C.1 and C.2

If the Required Action and associated Completion Time of Condition B cannot be met, operation in MODES 1 and 2 may continue provided power is promptly reduced to ≤ 70% RTP within 1 hour and the CEA is restored to within alignment limits in one additional hour. Operation with THERMAL POWER ≤ 70% RTP provides additional margin to offset the increase in F_r^T, thereby avoiding violation of the fuel design limits.

The Completion Time of Required Action C.1 provides the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System. The Completion Time of C.2 to restore the CEA to within alignment limits is sufficient to take appropriate corrective action to realign the CEA following the power reduction while recognizing the importance of minimizing the effects of xenon redistribution.

BASES

ACTIONS (continued)

~~B~~.1, ~~B~~.2.1, and ~~B~~.2.2

The CEA ~~motion inhibit~~ permits CEA motion within the requirements of LCO 3.1.6, "Regulating Control Element Assembly Insertion Limits," and prevents regulating CEAs from being misaligned from other CEAs in the group.

Performing SR 3.1.4.1 within 1 hour and every 4 hours thereafter, is considered acceptable in view of other information continuously available to the operator in the control room.

With the CEA ~~motion inhibit~~ inoperable, a Completion Time of 6 hours is allowed for restoring the CEA ~~motion inhibit~~ to OPERABLE status, or placing and maintaining the CEA drive switch in either the "off" or "manual" position~~.~~, fully withdrawing the CEAs in groups ~~3 and 4~~, and withdrawing all CEAs in group ~~5~~ to < 5% insertion~~.~~,

Placing the CEA drive switch in the "off" or "manual" position ensures the CEAs will not move ~~in response to Reactor Regulating System automatic motion commands~~. Withdrawal of the CEAs to the positions required in the Required Action ~~B~~.2.2 ensures that core perturbations in local burnup, peaking factors, and SDM will not be more adverse than the Conditions assumed in the safety analyses and LCO setpoint determination (Ref. ~~6~~).

The 6 hour Completion Time takes into account Required Action ~~B~~.1, the protection afforded by the CEA deviation circuits, and other information continuously available to the operator in the control room, so that during actual CEA motion, deviations can be detected.

Required Action ~~B~~.2.2 is modified by a Note indicating that this Required Action shall not be performed when in conflict with either Required Action ~~A~~.1, ~~A~~.2, or ~~C~~.1.

BASES

ACTIONS (continued)

~~C~~.1

When the CEA deviation circuit is inoperable, performing SR 3.1.4.1, within 1 hour and every 4 hours thereafter, ensures improper CEA alignments are identified before unacceptable flux distributions occur. The specified Completion Times take into account other information continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and the protection provided by the CEA inhibit and deviation circuit is not required.

~~D~~.1

If the Required Action or associated Completion Time of Condition A, ~~Condition B, or Condition~~ C is not met, one or more regulating or shutdown CEAs are inoperable, or two or more CEAs are misaligned by ~~>~~ [15 inches], the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside its MODE of applicability. Continued operation is not allowed in the case of more than one CEA misaligned from any other CEA in its group by ~~>~~ [15 inches], or one or more CEAs inoperable. This is because these cases are indicative of a loss of SDM and power distribution, and a loss of safety function, respectively.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification is required that individual CEA positions are within [~~7~~ inches] (indicated reed switch positions) of all other CEAs in the group within 1 hour of any CEA movement of > 7.5 inches. The CEA position verification after each movement of > 7.5 inches ensures that the CEAs in that group are properly aligned at the time when CEA misalignments are most likely to have occurred. ~~[The 12 hour Frequency allows the operator to detect a CEA that is beginning to deviate from its expected position. The specified Frequency takes into account other CEA position information that is continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA motion inhibit and deviation circuits.~~

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

6

SR 3.1.4.2

prevents any CEA from being misaligned from all other CEAs in its group by more than the LCO limit (indicated position), maintains the CEA group overlap and sequencing requirements of LCO 3.1.6, and prevents the regulating CEAs from being inserted beyond the Power Dependent Insertion Limit of COLR Figure 3.1-2

10

block circuit

Demonstrating the CEA ~~motion inhibit~~ OPERABLE verifies that the CEA ~~motion inhibit is functional~~, even if it is not regularly operated. ~~[The 92 day Frequency takes into account other information continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA deviation circuits.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

6

SR 3.1.4.3

Demonstrating the CEA deviation circuit is OPERABLE verifies the circuit is functional. ~~[The 92 day Frequency takes into account other information continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA motion inhibit.~~

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3

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.1.4.4

Verifying each CEA is trippable would require that each CEA be tripped. In MODES 1 and 2, tripping each CEA would result in radial or axial power tilts, or oscillations. Therefore, individual CEAs are exercised to provide increased confidence that all CEAs continue to be trippable, even if they are not regularly tripped. A movement of [~~5~~ inches] is adequate to demonstrate motion without exceeding the alignment limit when only one CEA is being moved. ~~[The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the CEAs.~~

7.5

6

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

Between required performances of SR 3.1.4.4, if a CEA(s) is discovered to be immovable, but remains trippable, the CEA is considered to be OPERABLE. At any time, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of the CEA(s) must be made, and appropriate action taken.

3

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.5

~~Performance of a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position over the entire length of the CEA's travel. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. [Since this Surveillance must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other surveillances being performed at shorter Frequencies, which determine the OPERABILITY of the CEA Reed Switch Indication System.~~

11

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~SR 3.1.4.6~~

and installation

3

Verification of CEA drop times determined that the maximum CEA drop time permitted is consistent with the assumed drop time used in that safety analysis (Ref. ~~7~~). Measuring drop times prior to reactor criticality, after reactor vessel head removal, ensures that reactor internals and CEDM will not interfere with CEA motion or drop time and that no degradation in these systems has occurred that would adversely affect CEA motion or drop time. Individual CEAs whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is

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The individual (shutdown and regulating) CEA drop time from the fully withdrawn position shall be determined from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches the 90% insertion position with Tavg greater than or equal to 515°F.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

performed prior to criticality, based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned unit transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR, ~~Section [ ]~~.

U

Chapter 15

~~4. FSAR, Section [ ].~~

~~5. FSAR, Section [ ].~~

~~6. FSAR, Section [ ].~~

~~7. FSAR, Section [ ].~~

6

3

3

1

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

1

B 3.1.4 Control Element Assembly (CEA) Alignment ~~(Analog)~~

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and regulating Control Element Assemblies (CEAs) is an initial assumption in all safety analyses that assume CEA insertion upon reactor trip. Maximum CEA misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10 and GDC 26 (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a CEA to become inoperable or to become misaligned from its group. CEA inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CEA worth for reactor shutdown. Therefore, CEA alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on CEA alignment and OPERABILITY have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CEAs are moved by their control element drive mechanisms (CEDMs). ~~Each CEDM moves its CEA~~ one step (approximately ¾ inch) at a time~~, but at varying rates (steps per minute) depending on the signal output from the Control Element Drive Mechanism Control System (CEDMCS)~~.

2

The CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating CEAs also provide reactivity (power level) control during normal operation and transients. ~~Their movement may be automatically controlled by the Reactor Regulating System.~~

2

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems, which are the ~~Plant Computer CEA~~ Position Indication System and the Reed Switch Position Indication System.

3

Pulse Counting

3

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BASES

BACKGROUND (continued)

Pulse Counting

The ~~Plant Computer CEA~~ Position Indication System counts the commands sent to the CEA gripper coils from the CEDM Control System that moves the CEAs. There is a one step counter for each ~~group of~~ CEA. Individual CEAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. ~~Plant Computer CEA~~ Position Indication System is considered highly precise (± 1 step or ± ¾ inch). ~~If a CEA does not move one step for each command signal, the step counter will still count the command and incorrectly reflect the position of the CEA~~.

3

An individual CEA may be moved in the Manual Individual Mode. The Pulse Counting

A step is counted after the "Step Complete" signal is received.

3 4

4 | R1

The Reed Switch Position Indication System provides a highly accurate indication of actual CEA position, but at a lower precision than the step counters. This system is based on ~~inductive analog~~ signals from a series of reed switch~~es~~ spaced ~~along a tube with a center to center distance of~~ 1.5 inch~~es~~, which is two steps. ~~To increase the reliability of the system, there are redundant reed switches at each position.~~

magnetically actuated

position transmitters

at

intervals

4

3

APPLICABLE SAFETY ANALYSES

CEA misalignment accidents are analyzed in the safety analysis (Ref. 3). The accident analysis defines CEA misoperation as an~~y~~ event~~, with the exception of sequential group withdraws,~~ which could result from a single malfunction in the reactivity control systems. ~~For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDMCS, or by operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper. Inadvertent withdrawal of a single CEA may be caused by the opening of the electrical circuit of the CEDM holding coil for a full length or part length CEA. A dropped CEA could be caused by an electrical failure in the CEA coil power programmers.~~

5

The acceptance criteria for addressing CEA inoperability or misalignment are that:

a. There shall be no violations of either:

   1. Specified acceptable fuel design limits or

   2. Reactor Coolant System (RCS) pressure boundary integrity and

b. The core must remain subcritical after accident transients.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~Three types of~~ misalignment are distinguished in the safety analysis (Ref. 1). ~~During movement of a group, one CEA may stop moving while the other CEAs in the group continue. This condition may cause excessive power peaking. The second type of~~ misalignment occurs if one CEA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining CEAs to meet the SDM requirement with the maximum worth CEA stuck fully withdrawn. If a CEA is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck CEAs into account. ~~The third type of~~ misalignment occurs when one CEA drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power, due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

~~Two types of~~ analyses are performed in regard to static CEA misalignment (Ref. ~~4~~). With CEA banks at their insertion limits, one type of analysis considers the case when any one CEA is inserted ~~[~~ ~~]~~ inches into the core. The second type of analysis considers the case of a single CEA withdrawn ~~[~~ ~~]~~ inches from a bank inserted into its insertion limit. Satisfying limits on departure from nucleate boiling ratio (DNBR) in both of these cases bounds the situation when a CEA is misaligned from its group by ~~[~~7 inches~~]~~.

Another type of misalignment occurs if one CEA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth CEA also fully withdrawn (Ref. ~~5~~).

Since the CEA drop incidents result in the most rapid approach to specified acceptable fuel design limits (SAFDLs) caused by a CEA misoperation, the accident analysis analyzed a single full length CEA drop. The most rapid approach to the DNBR SAFDL may be caused by a single full length CEA drop or a CEA subgroup drop, depending upon initial conditions.

~~All of the above CEA misoperations will result in an automatic reactor trip.~~ In the case of the full length CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The results of the CEA misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or RCS pressure occur.

CEA alignment limits and OPERABILITY requirements satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on shutdown and regulating CEA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on CEA OPERABILITY ensure that upon reactor trip, the CEAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The CEA OPERABILITY requirements (i.e., trippability) are separate from alignment requirements which ensure that the CEA banks maintain the correct power distribution and CEA alignment. The CEA OPERABILITY requirement is satisfied provided the CEA will fully insert in the required CEA drop time assumed in the safety analysis. CEA control malfunctions that result in the inability to move a CEA (e.g., CEA lift rod failures), but do not impact trippability, do not result in CEA inoperability.

The requirement is to maintain the CEA alignment to within ~~[~~7 inches~~]~~ between any CEA and its group. The minimum misalignment assumed in safety analysis is ~~[~~15 inches~~]~~, and in some cases, a total misalignment from fully withdrawn to fully inserted is assumed.

6

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMS, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on CEA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of CEAs have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the CEAs are bottomed, and the reactor is shut down and not producing fission power. In the shutdown Modes, the OPERABILITY of the shutdown and regulating CEAs has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

3

BASES

ACTIONS

A.1 ~~and A.2~~

If one or more CEAs (regulating or shutdown) are misaligned by > [7 inches] and ≤ [15 inches], ~~or one CEA is misaligned by > [15 inches], continued operation in MODES 1 and 2 may continue, provided, within 1 hour, the power is reduced to ≤ 70% RTP, and within~~ 2 hours ~~CEA alignment is restored~~. Regulating CEA alignment can be restored by either aligning the misaligned CEA(s) to within [7 inches] of its group or aligning the misaligned CEA's group to within [7 inches] of the misaligned CEA. Shutdown CEA alignment can be restored by aligning the misaligned CEA(s) to within [7 inches] of its group.

~~Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Reducing THERMAL POWER in accordance with Figure 3.1.4-1 (in the associated LCO) ensures acceptable power distributions are maintained (Ref. 6).~~ For ~~small misalignments~~ (~~<~~ [15 inches]) of the CEAs, there is:

a. A small effect on the time dependent long term power distributions relative to those used in generating LCOs and limiting safety system settings (LSSS) setpoints,

b. A negligible effect on the available SDM, and

c. A small effect on the ejected CEA worth used in the accident analysis.

~~With a large CEA misalignment (≥ [15 inches]), however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints. The effect on the available SDM and the ejected CEA worth used in the accident analysis remain small. Therefore, this condition is limited to a single CEA misalignment, while still allowing 2 hours for recovery.~~

~~In both cases, a~~ 2 hour~~s~~ ~~time period~~ is sufficient to:

a. Identify cause of a misaligned CEA, ~~and~~

b. Take appropriate corrective action to realign the CEAs~~, and~~

~~c. Minimize the effects of xenon redistribution~~.

Bases Insert 1

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BASES INSERT 1

B.1

If one CEA is misaligned from its group by > 15 inches, CEA alignment must be restored to ≤ 15 inches in 1 hour. Regulating CEA alignment can be restored by either aligning the misaligned CEA to ≤ 15 inches of its group or aligning the misaligned CEA's group to ≤ 15 inches of the misaligned CEA. Shutdown CEA alignment can be restored by aligning the misaligned CEA to ≤ 15 inches of its group.

Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. With a major CEA alignment deviation (> 15 inches), misalignment could cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints. The effect on the available SDM and the ejected CEA worth used in the accident analysis remain small. Therefore, this condition is limited to a single major CEA alignment deviation and the time constraints provided in the applicable COLR figure for recovery. A higher total integrated peaking factor (F_r^T) prior to the misalignment could result in a shorter time constraint, further limiting the Completion Time before requiring a power reduction.

The Completion Time in accordance with the COLR is dependent upon the F_r^T prior to the major CEA alignment deviation. A worst case analysis has shown that a DNBR SAFDL violation may occur during this condition. This potential DNBR SAFDL violation is eliminated by limiting the time operation is permitted at full power before a power reduction is required.

C.1 and C.2

If the Required Action and associated Completion Time of Condition B cannot be met, operation in MODES 1 and 2 may continue provided power is promptly reduced to ≤ 70% RTP within 1 hour and the CEA is restored to within alignment limits in one additional hour. Operation with THERMAL POWER ≤ 70% RTP provides additional margin to offset the increase in F_r^T, thereby avoiding violation of the fuel design limits.

The Completion Time of Required Action C.1 provides the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System. The Completion Time of C.2 to restore the CEA to within alignment limits is sufficient to take appropriate corrective action to realign the CEA following the power reduction while recognizing the importance of minimizing the effects of xenon redistribution.

BASES

ACTIONS (continued)

~~B~~.1, ~~B~~.2.1, and ~~B~~.2.2

The CEA ~~motion inhibit~~ permits CEA motion within the requirements of LCO 3.1.6, "Regulating Control Element Assembly Insertion Limits," and prevents regulating CEAs from being misaligned from other CEAs in the group.

Performing SR 3.1.4.1 within 1 hour and every 4 hours thereafter, is considered acceptable in view of other information continuously available to the operator in the control room.

With the CEA ~~motion inhibit~~ inoperable, a Completion Time of 6 hours is allowed for restoring the CEA ~~motion inhibit~~ to OPERABLE status, or placing and maintaining the CEA drive switch in either the "off" or "manual" position, fully withdrawing the CEAs in groups 3 and 4, and withdrawing all CEAs in group 5 to < ~~5~~% insertion~~.~~,

Placing the CEA drive switch in the "off" or "manual" position ensures the CEAs will not move ~~in response to Reactor Regulating System automatic motion commands~~. Withdrawal of the CEAs to the positions required in the Required Action ~~B~~.2.2 ensures that core perturbations in local burnup, peaking factors, and SDM will not be more adverse than the Conditions assumed in the safety analyses and LCO setpoint determination (Ref. ~~6~~).

The 6 hour Completion Time takes into account Required Action ~~B~~.1, the protection afforded by the CEA deviation circuits, and other information continuously available to the operator in the control room, so that during actual CEA motion, deviations can be detected.

Required Action ~~B~~.2.2 is modified by a Note indicating that this Required Action shall not be performed when in conflict with either Required Action ~~A~~.1, ~~A~~.2, or ~~C~~.1.

1

BASES

ACTIONS (continued)

E → ~~C~~.1

8

When the CEA deviation circuit is inoperable, performing SR 3.1.4.1, within 1 hour and every 4 hours thereafter, ensures improper CEA alignments are identified before unacceptable flux distributions occur. The specified Completion Times take into account other information continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and the protection provided by the CEA inhibit and deviation circuit is not required.

F → ~~D~~.1

8

If the Required Action or associated Completion Time of Condition A, ~~Condition B, or Condition~~ C, D, or E is not met, one or more regulating or shutdown CEAs are inoperable, or two or more CEAs are misaligned by > ~~[~~15 inches~~]~~, the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside its MODE of applicability. Continued operation is not allowed in the case of more than one CEA misaligned from any other CEA in its group by > ~~[~~15 inches~~]~~, or one or more CEAs inoperable. This is because these cases are indicative of a loss of SDM and power distribution, and a loss of safety function, respectively.

6

6

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification is required that individual CEA positions are within ~~[~~7 inches~~]~~ (indicated reed switch positions) of all other CEAs in the group within 1 hour of any CEA movement of > 7~~.5~~ inches. The CEA position verification after each movement of > 7~~.5~~ inches ensures that the CEAs in that group are properly aligned at the time when CEA misalignments are most likely to have occurred. ~~[The 12 hour Frequency allows the operator to detect a CEA that is beginning to deviate from its expected position. The specified Frequency takes into account other CEA position information that is continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA motion inhibit and deviation circuits.~~

9

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3

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

6

SR 3.1.4.2

prevents any CEA from being misaligned from all other CEAs in its group by more than the LCO limit (indicated position), maintains the CEA group overlap and sequencing requirements of LCO 3.1.6, and prevents the regulating CEAs from being inserted beyond the Power Dependent Insertion Limit of COLR Figure 3.1-2

10

block circuit

Demonstrating the CEA ~~motion inhibit~~ OPERABLE verifies that the CEA ~~motion inhibit is functional~~, even if it is not regularly operated. ~~[The 92 day Frequency takes into account other information continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA deviation circuits.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

6

SR 3.1.4.3

Demonstrating the CEA deviation circuit is OPERABLE verifies the circuit is functional. ~~[The 92 day Frequency takes into account other information continuously available to the operator in the control room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA motion inhibit.~~

6

3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.1.4.4

Verifying each CEA is trippable would require that each CEA be tripped. In MODES 1 and 2, tripping each CEA would result in radial or axial power tilts, or oscillations. Therefore, individual CEAs are exercised to provide increased confidence that all CEAs continue to be trippable, even if they are not regularly tripped. A movement of [~~5~~ inches] is adequate to demonstrate motion without exceeding the alignment limit when only one CEA is being moved. ~~[The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the CEAs.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

Between required performances of SR 3.1.4.4, if a CEA(s) is discovered to be immovable, but remains trippable, the CEA is considered to be OPERABLE. At any time, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of the CEA(s) must be made, and appropriate action taken.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.5

~~Performance of a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position over the entire length of the CEA's travel. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. [Since this Surveillance must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other surveillances being performed at shorter Frequencies, which determine the OPERABILITY of the CEA Reed Switch Indication System.~~

11

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~SR 3.1.4.6~~

and installation

3

Verification of CEA drop times determined that the maximum CEA drop time permitted is consistent with the assumed drop time used in that safety analysis (Ref. ~~7~~). Measuring drop times prior to reactor criticality, after reactor vessel head removal, ensures that reactor internals and CEDM will not interfere with CEA motion or drop time and that no degradation in these systems has occurred that would adversely affect CEA motion or drop time. Individual CEAs whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is

3

The individual (shutdown and regulating) CEA drop time from the fully withdrawn position shall be determined from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches the 90% insertion position with Tavg greater than or equal to 515°F.

12

3

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

performed prior to criticality, based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned unit transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.

2. 10 CFR 50.46.

U

Chapter 15

3. FSAR, ~~Section [ ]~~.

~~4. FSAR, Section [ ].~~

~~5. FSAR, Section [ ].~~

~~6. FSAR, Section [ ].~~

~~7. FSAR, Section [ ].~~

6

3

3

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.4, BASES, CONTROL ELEMENT ASSEMBLY (CEA) ALIGNMENT

1. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. The ISTS Bases states that the CEAs may be moved at varying rates (steps per minute), and that the CEA movement may be automatically controlled by the Reactor Regulating System. Moving CEA(s) at varying rates and automatic control of CEA(s) is not a design feature for PSL Unit 1 and Unit 2 due to the incorporation of the ARCH (Advanced Rod Control Hybrid) system that controls CEA movement. Therefore, the associated text is deleted.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

4. ISTS Bases is changed to incorporate ARCH (Advanced Rod Control Hybrid) system design features installed at PSL Unit 1 and PSL Unit 2. For ARCH, an individual CEA may be moved in the Manual Individual Mode, therefore, the description is added. ISTS states that if a CEA does not move one step for each command signal, the step counter will still count the command and incorrectly reflect the position of the CEA. For ARCH, a step is counted only after the Step Complete signal is received. The associated text is changed. The ISTS states that the system is based on analog signals. For ARCH, the system uses a resistance voltage divider network. Therefore, the associated text “inductive analog” is deleted. The ISTS states that during movement of a group, one CEA may stop moving while the other CEAs in the group continue. This condition may cause excessive power peaking. For ARCH, this is not a valid failure mode. Therefore, this text is deleted.

5. ISTS includes numerous examples of failure modes for CEA alignment accidents. These examples are unnecessary information in the Applicable Safety Analysis and are deleted. The deleted examples do not affect the subsequent description for the Applicable Safety Analysis.

6. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

7. The ISTS alignment limits are changed to the PSL Unit 1 current licensing basis. PSL Unit 2 current licensing basis is also retained and matches the ISTS without change.

8. ACTIONS A and B are added to ITS 3.1.4 and ISTS 3.1.4, Condition A (ITS 3.1.4, Condition C) is modified to reflect PSL CTS 3.1.3.1 Actions as modified by the CTS Discussion of Changes. The ISTS Bases changes reflect the changes associated with the ISTS Specification.

9. Changes made to the PSL ITS Bases to correct a grammatical error in the ISTS Bases or align the text with the PSL ITS Specification.

10. Surveillance Requirement details removed from CTS 4.1.3.1.3 and CTS 4.1.3.1.4 and relocated to the ITS Bases of SR 3.1.4.2. See CTS DOC LA03.

11. ISTS SR 3.1.4.5 is deleted and subsequent Surveillance is renumbered. PSL developed ITS 3.1.7, Control Element Assembly (CEA) Position Indication, and requires performance of ITS SR 3.1.7.1 to determine OPERABILITY of the reed switch position indicating channels and the pulse counting position indicating channels in accordance with the Surveillance Frequency Control Program.

12. CTS 3.1.3.4 requires the individual (shutdown and regulating) CEA drop time from the fully withdrawn position shall be less than or equal to 3.1 seconds (PSL Unit 1) and less than or equal to 3.25 seconds (PSL Unit 2) from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches the 90% insertion position with Tavg greater than or equal to 515°F. These details for the CEA drop time surveillance are relocated to the ITS 3.1.4 Bases. See DOC LA04.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.4, CONTROL ELEMENT ASSEMBLY (CEA) ALIGNMENT

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

3.1.5 Shutdown Control Element Assembly (CEA) Insertion Limits

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTIVITY CONTROL SYSTEMS

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

LCO 3.1.5 ~~3.1.3.5~~ All shutdown CEAs shall be withdrawn to ~~at least~~ 129~~.0~~ inches.

Applicability **APPLICABILITY**: MODE~~S~~ 1 ~~and~~ 2~~*#~~.

A02 M01

ACTION:

ACTION A ~~With a maximum of~~ one shutdown CEA ~~withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches,~~ within ~~one~~ hour ~~either:~~

Applicability Note

L01

Required Action A.1 ~~a. Withdraw the~~ CEA to ~~at least 129.0 inches, or~~

Required Action B.1 ~~b. Declare the CEA inoperable and apply Specification 3.1.3.1.~~

L02

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1 ~~4.1.3.5~~ ~~Each~~ shutdown CEA ~~shall be determined to be~~ withdrawn ~~to at least~~ 129~~.0~~ inches~~:~~

Applicability MODE 2 ~~a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and~~

L03

SR 3.1.5.1 Frequency ~~b.~~ In accordance with the Surveillance Frequency Control Program.

~~* See Special Test Exception 3.10.2.~~

A02

~~# With K_{eff} ≥ 1.0.~~

M01

REACTIVITY CONTROL SYSTEMS

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

LCO 3.1.5 ~~3.1.3.5~~ All shutdown CEAs shall be withdrawn to ~~greater than or equal to~~ ≥ 129.0 inches.

Applicability **APPLICABILITY**: MODE~~S~~ 1 ~~and~~ 2~~*~~, with any regulating CEA not fully inserted~~#~~. A02 M01

ACTION:

ACTION A ~~With a maximum of~~ one or more shutdown CEA~~s~~ not within limit. ~~withdrawn to less than 129.0 inches,~~ ~~except~~ This LCO is not applicable while performing SR 3.1.4.4. L01

Applicability Note ~~for surveillance testing pursuant to Specification 4.1.3.1.2,~~ within ~~1~~ 2 hour within limit. L01

~~either:~~

Required Action A.1 ~~a.~~ ~~Withdraw the~~ Restore shutdown CEA(s) to ~~greater than or equal to~~ within limit. 129.0 inches, ~~or~~

Required Action B.1 ~~b.~~ ~~Declare the CEA inoperable and apply Specification 3.1.3.1.~~ Be in MODE 3 within 6 hours. L02

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1 ~~4.1.3.5~~ Verify Each shutdown CEA ~~shall be determined to be~~ is withdrawn to ~~greater than or equal to~~ ≥ 129.0 inches.

Applicability MODE 2 ~~a.~~ ~~Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and~~ L03

SR 3.1.5.1 Frequency ~~b.~~ In accordance with the Surveillance Frequency Control Program thereafter.

~~* See Special Test Exception 3.10.2.~~ A02

~~# With K_{eff} greater than or equal to 1.0.~~ M01

DISCUSSION OF CHANGES
ITS 3.1.5, SHUTDOWN CONTROL ELEMENT ASSEMBLY (CEA) INSERTION LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.1.3.5 Applicability is MODES 1 and 2 with a footnote (footnote *) for MODE 2 stating "See Special Test Exception 3.10.2," and a footnote (footnote #) for MODE 2 stating "With $K_{eff} \geq 1.0$." ITS 3.1.5 does not contain the footnotes or a reference to the Special Test Exception or reference to the K_{eff}. This changes the CTS by not including footnote * and footnote # in the ITS.

The purpose of the footnote references is to alert the user that conditions exist that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS. See DOC M01 for additional discussion of change for the footnote # for MODE 2 stating "With $K_{eff} \geq 1.0$."

MORE RESTRICTIVE CHANGES

M01 CTS 3.1.3.5 is applicable in MODES 1 and 2 with $k_{eff} \geq 1.0$. MODE 2 is modified by CTS 3.1.3.5 footnote #. ITS 3.1.5 is applicable in MODES 1 and 2. This changes the CTS by expanding the Applicability from MODE 2 with the reactor critical to all of MODE 2.

The purpose of CTS 3.1.3.5 is to ensure that the shutdown CEAs are fully withdrawn prior to withdrawing the regulating CEAs to ensure that there is sufficient shutdown margin available to quickly shutdown the reactor. This change is acceptable because applying the requirement prior to moving the CEAs and bringing the reactor critical ensures that the shutdown margin is available and is consistent with plant operation, in that the shutdown CEAs are withdrawn to at least 129.0 inches before beginning to withdraw the regulating CEAs and approaching criticality. This change is designated as more restrictive because it increases the conditions under which Technical Specification controls will be applied.

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS 3.1.5, SHUTDOWN CONTROL ELEMENT ASSEMBLY (CEA) INSERTION LIMITS

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 (*Category 4 – Relaxation of Required Action)* CTS 3.1.3.5 ACTION states that with one shutdown CEA withdrawn to less than 129.0 inches, then withdraw the CEA to at least 129.0 inches within 1 hour. ITS 3.1.5 Required Action A.1 requires that the shutdown CEA be withdrawn to at least 129.0 inches within 2 hours. This changes the CTS by changing the Completion Time from one hour to two hours.

The purpose of CTS 3.1.3.5 ACTION is to restore a shutdown CEA not withdrawn to at least 129.0 inches, to within limits within 1 hour. Prior to entering this Condition, the shutdown CEAs were fully withdrawn. If a shutdown CEA is then inserted into the core, its potential negative reactivity is added to the core as it is inserted. The 2 hour Completion Time allows the operator adequate time to adjust the CEA(s) in an orderly manner and is consistent with the Completion Times in LCO 3.1.4, “Control Element Assembly (CEA) Alignment.” This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, the capacity and capability of remaining features, and the low probability of a DBA occurring during the allowed Completion Time. The Completion Time of 2 hours gives the operator sufficient time to adjust the CEA(s) in an orderly manner. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.5 ACTION a. and ACTION B provide compensatory actions for a maximum of one shutdown CEA withdrawn to less than 129.0 inches. The actions require within one hour either restore the CEA to at least 129.0 inches or declare the CEA inoperable and apply ACTION Specification 3.1.3.1. For more than one shutdown CEA not withdrawn to at least 129.0 inches, CTS 3.1.3.5 does not contain a specific requirement; therefore, entry into CTS 3.0.3 is required. ITS 3.1.5 ACTION A and ACTION B provide Required Actions for one or more shutdown CEAs not within limits. ITS 3.1.5 Required Action A.1 requires restoring the shutdown CEA(s) to within the limit within 2 hours. Additionally, ITS 3.1.5 Required Action B.1 requires if any Required Action and associated Completion Time is not met, the unit must be in MODE 3 within 6 hours. This changes the CTS by allowing more than one shutdown CEA to be outside the limits specified in the LCO, eliminates the allowance to declare the CEA inoperable and to take the ACTIONS of Specification 3.1.3.1, and eliminates the requirement to enter CTS 3.0.3 if more than one shutdown CEA is not within the limits specified in the LCO, or if any Required Action and associated Completion Time is not met.

The purpose of CTS 3.1.3.5 ACTION a. is to ensure the shutdown CEAs are withdrawn to at least 129.0 inches to ensure that there is sufficient margin available to quickly shutdown the reactor. This change is acceptable because

the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering that only a small amount of time is provided to establish the required features, and the low probability of a DBA occurring during the repair period. Allowing an additional hour to restore one or more shutdown CEAs to within the limit is appropriate allows the operator adequate time to adjust the CEA(s) in an orderly manner and is consistent with the Completion Times in LCO 3.1.4, "Control Element Assembly (CEA) Alignment." .

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.3.5.a requires verification that each shutdown CEA is within the limit specified in the LCO within 15 minutes prior to withdrawal of any CEAs in the regulating groups during an approach to reactor criticality. ITS 3.1.5 does not require verification that each shutdown CEA is within the limit specified in the LCO within 15 minutes prior to withdrawal of any CEAs in the regulating groups during an approach to reactor criticality. This changes the CTS by eliminating the requirement that each CEA is within the limit specified in the LCO within 15 minutes prior to withdrawal of any CEAs in the regulating groups during an approach to reactor criticality.

The purpose of CTS 4.1.3.5.a is to verify the shutdown CEAs are withdrawn to at least 129.0 inches prior to withdrawing the regulating CEAs. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify the equipment being used to meet the LCO can perform its required function. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence the equipment can perform its assumed safety function. Under the ITS Applicability of MODE 2 and the requirement of ITS LCO 3.0.4, the shutdown CEAs must be withdrawn to at least 129.0 inches prior to entering the ITS Applicability of MODE 2. However, it is not required to verify compliance within a specified time prior to initial withdrawal of regulating CEAs. Specifying a time is not necessary to ensure the shutdown CEAs are withdrawn to at least 129.0 inches prior to initial withdrawal of regulating CEA withdrawal if the shutdown CEAs are withdrawn to at least 129.0 before withdrawing the regulating CEAs.

This change is designated as less restrictive because a Surveillance which was required in CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~ 3

3.1.5 Shutdown Control Element Assembly (CEA) Insertion Limits ~~(Analog)~~

LCO 3.1.3.5 LCO 3.1.5 All shutdown CEAs shall be withdrawn to ≥ ~~[~~129~~]~~ inches. 2

Applicability APPLICABILITY: MODE 1,
MODE 2 with any regulating CEA not fully inserted.

Action ----------NOTE----------
This LCO is not applicable while performing SR 3.1.4.4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a. A. One or more shutdown CEAs not within limit.	A.1 Restore shutdown CEA(s) to within limit.	2 hours
Action b. B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 4.1.3.5 SR 3.1.5.1	Verify each shutdown CEA is withdrawn ≥ ~~[~~129~~]~~ inches.	~~[12 hours~~ ~~OR~~ SR 4.1.3.5.b In accordance with the Surveillance Frequency Control Program ~~]~~ 2

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~ 3

3.1.5 Shutdown Control Element Assembly (CEA) Insertion Limits ~~(Analog)~~

LCO 3.1.3.5

LCO 3.1.5 All shutdown CEAs shall be withdrawn to ≥ ~~[~~129~~]~~ inches. 2

Applicability

APPLICABILITY: MODE 1,
MODE 2 with any regulating CEA not fully inserted.

Action

----------NOTE----------
This LCO is not applicable while performing SR 3.1.4.4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a. A. One or more shutdown CEAs not within limit.	A.1 Restore shutdown CEA(s) to within limit.	2 hours
Action b. B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 4.1.3.5 SR 3.1.5.1	Verify each shutdown CEA is withdrawn ≥ ~~[~~129~~]~~ inches.	~~[12 hours~~ ~~OR~~ SR 4.1.3.5.b In accordance with the Surveillance Frequency Control Program ~~]~~ 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.5, SHUTDOWN CONTROL ELEMENT ASSEMBLY (CEA) INSERTION LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~ 4

B 3.1.5 Shutdown Control Element Assembly (CEA) Insertion Limits ~~(Analog)~~

BASES

BACKGROUND

The insertion limits of the shutdown Control Element Assemblies (CEAs) are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected CEA worth, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on shutdown CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected CEA worth, and SDM limits are preserved.

The shutdown CEAs are arranged into groups that are radially symmetric. Therefore, movement of the shutdown CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip.

The design calculations are performed with the assumption that the shutdown CEAs are withdrawn prior to the regulating CEAs. The shutdown CEAs can be fully withdrawn without the core going critical. This provides available negative reactivity for SDM in the event of boration errors. The shutdown CEAs are controlled manually ~~or automatically~~ by the control room operator. During normal unit operation, the shutdown CEAs are fully withdrawn. The shutdown CEAs must be completely withdrawn from the core prior to withdrawing any regulating CEAs during an approach to criticality. The shutdown CEAs are then left in this position until the reactor is shut down. They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal. 3

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown CEAs are fully withdrawn any time the reactor is critical. This ensures that:

a. The minimum SDM is maintained and

b. The potential effects of a CEA ejection accident are limited to acceptable limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

CEAs are considered fully withdrawn at 129 inches, since this position places them outside the active region of the core.

On a reactor trip, all CEAs (shutdown and regulating), except the most reactive CEA, are assumed to insert into the core. The shutdown and regulating CEAs shall be at their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating CEAs may be partially inserted in the core as allowed by LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits." The shutdown CEA insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of regulating CEAs and shutdown CEAs (less the most reactive CEA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown CEA insertion limit also limits the reactivity worth of an ejected shutdown CEA.

The acceptance criteria for addressing shutdown CEA as well as regulating CEA insertion limits and inoperability or misalignment are that:

a. There be no violation of either:

   1. Specified acceptable fuel design limits or

   2. Reactor Coolant System pressure boundary damage and

b. The core remains subcritical after accident transients.

As such, the shutdown CEA insertion limits affect safety analyses involving core reactivity, ejected CEA worth, and SDM (Ref. 3).

The shutdown CEA insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shutdown CEAs must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

BASES

APPLICABILITY

The shutdown CEAs must be within their insertion limits, with the reactor in MODES 1 and 2. The Applicability in MODE 2 begins anytime any regulating CEA is not fully inserted. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODE 3, 4, 5, or 6, the shutdown CEAs are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.4. This SR verifies the freedom of the CEAs to move, and requires the shutdown CEAs to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1

Prior to entering this condition, the shutdown CEAs were fully withdrawn. If a shutdown CEA(s) is then inserted into the core, its potential negative reactivity is added to the core as it is inserted.

~~If the CEA(s) is not restored to within limits within 1 hour, then an additional 1 hour is allowed for restoring the CEA(s) to within limits.~~ The 2 hour total Completion Time allows the operator adequate time to adjust the CEA(s) in an orderly manner and is consistent with the required Completion Times in LCO 3.1.4, "Control Element Assembly (CEA) Alignment." 1

B.1

When Required Action A.1 ~~or A.2~~ cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. 1

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown CEAs are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown CEAs will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown CEAs are withdrawn before the regulating CEAs are withdrawn during a unit startup.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[Since the shutdown CEAs are positioned manually by the control room operator, verification of shutdown CEA position at a Frequency of 12 hours is adequate to ensure that the shutdown CEAs are within their insertion limits. Also, the 12 hour Frequency takes into account other information available to the operator in the control room for the purpose of monitoring the status of the shutdown CEAs.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.

2. 10 CFR 50.46.

3. FSAR, Section ~~[~~ U 4.2 ~~]~~. 2

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~ 4

B 3.1.5 Shutdown Control Element Assembly (CEA) Insertion Limits ~~(Analog)~~

BASES

BACKGROUND

The insertion limits of the shutdown Control Element Assemblies (CEAs) are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected CEA worth, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on shutdown CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected CEA worth, and SDM limits are preserved.

The shutdown CEAs are arranged into groups that are radially symmetric. Therefore, movement of the shutdown CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip.

The design calculations are performed with the assumption that the shutdown CEAs are withdrawn prior to the regulating CEAs. The shutdown CEAs can be fully withdrawn without the core going critical. This provides available negative reactivity for SDM in the event of boration errors. The shutdown CEAs are controlled manually ~~or automatically~~ by the control room operator. During normal unit operation, the shutdown CEAs are fully withdrawn. The shutdown CEAs must be completely withdrawn from the core prior to withdrawing any regulating CEAs during an approach to criticality. The shutdown CEAs are then left in this position until the reactor is shut down. They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal. 3

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown CEAs are fully withdrawn any time the reactor is critical. This ensures that:

a. The minimum SDM is maintained and

b. The potential effects of a CEA ejection accident are limited to acceptable limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

CEAs are considered fully withdrawn at 129 inches, since this position places them outside the active region of the core.

On a reactor trip, all CEAs (shutdown and regulating), except the most reactive CEA, are assumed to insert into the core. The shutdown and regulating CEAs shall be at their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating CEAs may be partially inserted in the core as allowed by LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits." The shutdown CEA insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of regulating CEAs and shutdown CEAs (less the most reactive CEA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown CEA insertion limit also limits the reactivity worth of an ejected shutdown CEA.

The acceptance criteria for addressing shutdown CEA as well as regulating CEA insertion limits and inoperability or misalignment are that:

a. There be no violation of either:

   1. Specified acceptable fuel design limits or

   2. Reactor Coolant System pressure boundary damage and

b. The core remains subcritical after accident transients.

As such, the shutdown CEA insertion limits affect safety analyses involving core reactivity, ejected CEA worth, and SDM (Ref. 3).

The shutdown CEA insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shutdown CEAs must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

BASES

APPLICABILITY

The shutdown CEAs must be within their insertion limits, with the reactor in MODES 1 and 2. The Applicability in MODE 2 begins anytime any regulating CEA is not fully inserted. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODE 3, 4, 5, or 6, the shutdown CEAs are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.4. This SR verifies the freedom of the CEAs to move, and requires the shutdown CEAs to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1

Prior to entering this condition, the shutdown CEAs were fully withdrawn. If a shutdown CEA(s) is then inserted into the core, its potential negative reactivity is added to the core as it is inserted.

~~If the CEA(s) is not restored to within limits within 1 hour, then an additional 1 hour is allowed for restoring the CEA(s) to within limits.~~ The 2 hour total Completion Time allows the operator adequate time to adjust the CEA(s) in an orderly manner and is consistent with the required Completion Times in LCO 3.1.4, "Control Element Assembly (CEA) Alignment." 1

B.1

When Required Action A.1 ~~or A.2~~ cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. 1

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown CEAs are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown CEAs will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown CEAs are withdrawn before the regulating CEAs are withdrawn during a unit startup.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[Since the shutdown CEAs are positioned manually by the control room operator, verification of shutdown CEA position at a Frequency of 12 hours is adequate to ensure that the shutdown CEAs are within their insertion limits. Also, the 12 hour Frequency takes into account other information available to the operator in the control room for the purpose of monitoring the status of the shutdown CEAs.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.

2. 10 CFR 50.46.

3. FSAR, Section ~~[~~ U 4.2 ~~]~~. 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.5, BASES, SHUTDOWN CONTROL ELEMENT ASSEMBLY (CEA)
INSERTION LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS Bases states that the shutdown CEAs are controlled manually or automatically by the control room operator. Automatic control of CEAs is not a design feature for PSL Unit 1 and Unit 2. Therefore, the text “or automatic” is deleted.

4. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.5, SHUTDOWN CONTROL ELEMENT ASSEMBLY (CEA) INSERTION LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

3.1.6 Regulating Control Element Assembly (CEA) Insertion Limits

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTIVITY CONTROL SYSTEMS

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

power dependent insertion limit (PDIL) alarm circuit shall be OPERABLE, and the

A04

LCO 3.1.6 ~~3.1.3.6~~ The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits specified in the COLR ~~(regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches) with CEA insertion~~ between

LA01

Condition B and C the Long Term Steady State Insertion Limit~~s~~ and the Power Dependent Insertion Limit~~s~~ ~~restricted to:~~

Regulating CEA groups inserted

Condition B ~~a.~~ ≤ 4 hours per 24 hour interval~~,~~

>

Condition C ~~b.~~ ≤ 5 Effective Full Power Days per 30 ~~Effective Full Power Day~~ interval~~, and~~

(EFPD) EFPD

EFPD

365 EFPD

Condition C ~~c.~~ ≤ 14 ~~Effective Full Power Days~~ per ~~calendar year~~.

Applicability **APPLICABILITY**: MODES 1~~*~~ and 2~~*#~~.

M01

A02

ACTION:

Condition A ~~a.~~ ~~With the~~ regulating CEA groups inserted beyond the Power Dependent Insertion Limit~~s,~~ ~~except for surveillance testing pursuant to Specification~~ ~~4.1.3.1.2,~~ within two hours ~~either:~~

Applicability Note — This LCO is not applicable while performing SR 3.1.4.4.

Required Action A.1 ~~1.~~ Restore ~~the~~ regulating CEA groups to within ~~the~~ limits~~, or~~

the

Required Action A.2 ~~2.~~ Reduce THERMAL POWER to less than or equal to ~~that~~ fraction of

RTP ~~RATED THERMAL POWER which is~~ allowed by the CEA group position and insertion limits ~~specified~~ in the COLR.

ACTION B ~~b.~~ ~~With the~~ regulating CEA groups inserted between the Long Term Steady State Insertion Limit~~s~~ and the Power Dependent Insertion Limit~~s~~ for ~~intervals~~ > 4 hours per 24 hour interval~~, except during operation pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1, operation may proceed provided either:~~

A03

within 15 minutes

L01

Verify

Required Action B.1 ~~1.~~ ~~The~~ Short Term Steady State Insertion Limits are not exceeded~~, or~~

Restrict s

Required Action B.2 ~~2.~~ ~~Any subsequent~~ increase in THERMAL POWER ~~is restricted~~ to ≤ 5% ~~of~~

RTP ~~RATED THERMAL POWER~~ per hour.

~~* See Special Test Exception 3.10.2 and 3.10.5.~~

A02

~~# With K_{eff} ≥ 1.0.~~

M01

REACTIVITY CONTROL SYSTEMS

REGULATING CEA INSERTION LIMITS (Continued)

LIMITING CONDITION FOR OPERATION

ACTION C

c. ~~With the~~ regulating CEA groups inserted between the Long Term Steady State Insertion Limit~~s~~ and the Power Dependent Insertion Limit~~s~~ for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per ~~calendar year, except during operations pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1, either~~:

effective full power days ( ) 365 EFPD.

A03

Required Action C.1

1. Restore ~~the~~ regulating CEA groups to within ~~the Long Term Steady State Insertion~~ Limits within two hours~~, or~~

Required Action E.1

2. Be in ~~HOT STANDBY~~ MODE 3 within 6 hours.

SR 3.1.6.1 NOTE

NOTE Not required to be performed until 12 hours after entry into MODE 2.

L02

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

~~4.1.3.6~~ ~~The position of~~ Verify each regulating CEA group position ~~shall be determined to be~~ is within its ~~the Power Dependent~~ Insertion Limits in accordance with the Surveillance Frequency Control Program ~~except during time intervals when the~~ PDIL ~~Auctioneer~~ Alarm Circuit ~~is~~ inoperable~~, then verify the individual CEA positions at least~~ once per 4 hours. The accumulated times during which the regulating CEA groups are inserted ~~between~~ the ~~Long Term~~ Steady State Insertion Limits ~~and the~~ Power Dependent Insertion Limits ~~shall be determined~~ in accordance with the Surveillance Frequency Control Program.

ACTION D
Required Action D.1 thereafter

SR 3.1.6.2 Verify beyond

but within the

1 hour AND

Perform SR 3.1.6.1.

M02

SR 3.1.6.3

Verify PDIL alarm circuit is OPERABLE in accordance with the Surveillance Frequency Control Program.

A04

~~DELETED~~

ITS

ITS 3.1.6

A01

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

See ITS 3.1.1

SHUTDOWN MARGIN - T_{avg} > 200 °F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

Applicability

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN not within limits immediately initiate and continue boration at ≥ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

See ITS 3.1.4

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

SR 3.1.6.1

b. ~~When in MODES 1 or 2#,~~ in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

~~c. When in MODE 2## at least once during CEA withdrawal and in accordance with the Surveillance Frequency Control Program until the reactor is critical.~~

L03

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

See ITS 3.1.1

~~* See Special Test Exception 3.10.1.~~

~~# With K_{eff} ≥ 1.0.~~

~~## With K_{eff} < 1.0.~~

A02

3/4.2 POWER DISTRIBUTION LIMITS

See ITS 3.2.1

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent control limits of COLR Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

b. Be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or wi th the incore detector monitoring system.

4.2.1.3 Excore Detector Monitoring System – The excore detector monitoring system may be used for monitoring the linear heat rate by:

SR 3.1.6.1
SR 3.1.6.2

a. Verifying in accordance with the Surveillance Frequency Control Program that the ~~full length~~ CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.

A05

b. Verifying in accordance with the Surveillance Frequency Control Program that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on COLR Figure 3.2-2.

REACTIVITY CONTROL SYSTEMS

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

power dependent insertion limit (PDIL) alarm circuit shall be OPERABLE, and the

A04

LCO 3.1.6

~~3.1.3.6~~ The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits ~~shown on~~ COLR ~~Figure 3.1-2 (regulating CEAs are considered to be fully withdrawn in accordance with COLR Figure 3.1-2 when withdrawn to greater than or equal to 129.0 inches), with CEA insertion~~ between the Long Term Steady State Insertion Limit~~s~~ and the Power Dependent Insertion Limit~~s restricted to:~~

specified in the

LA01

Condition B and C

Regulating CEA groups inserted

Condition B

~~a.~~ ~~Less than or equal to~~ 4 hours per 24 hour interval,

>

(EFPD)

EFPD

Condition C

~~b.~~ ~~Less than or equal to~~ 5 Effective Full Power Days per 30 ~~Effective Full Power Day~~ interval~~, and~~

EFPD

365 EFPD

Condition C

~~c.~~ ~~Less than or equal to~~ 14 ~~Effective Full Power Days~~ per ~~calendar year~~.

Applicability

APPLICABILITY: MODES 1~~*~~ and 2~~*#~~.

M01

A02

ACTION:

Condition A

~~a.~~ ~~With the~~ regulating CEA groups inserted beyond the Power Dependent Insertion Limit~~s, except for surveillance testing pursuant to Specification 4.1.3.1.2,~~ within 2 hours ~~either:~~

Applicability Note

This LCO is not applicable while performing SR 3.1.4.4.

Required Action A.1

~~1.~~ Restore ~~the~~ regulating CEA groups to within ~~the~~ limits~~, or~~

the

Required Action A.2

~~2.~~ Reduce THERMAL POWER to less than or equal to ~~that~~ fraction of ~~RATED THERMAL POWER which is~~ allowed by the CEA group position and insertion limits ~~specified~~ in the COLR.

RTP

ACTION B

~~b.~~ ~~With the~~ regulating CEA groups inserted between the Long Term Steady State Insertion Limit~~s~~ and the Power Dependent Insertion Limit~~s~~ for ~~intervals greater than~~ 4 hours per 24 hour interval~~, operation may proceed provided either:~~

within 15 minutes

L01

Verify

Required Action B.1

~~1.~~ ~~The~~ Short Term Steady State Insertion Limits are not exceeded~~, or~~

Restrict

s

Required Action B.2

~~2.~~ ~~Any subsequent~~ increase in THERMAL POWER ~~is restricted to less than or equal~~ to 5% ~~of RATED THERMAL POWER~~ per hour.

≤

RTP

~~* See Special Test Exception 3.10.2, 3.10.4 and 3.10.5.~~

A02

~~# With K_{eff} greater than or equal to 1.0.~~

M01

REACTIVITY CONTROL SYSTEMS

ACTION: (Continued)

ACTION C

c. ~~With the~~ regulating CEA groups inserted between the Long Term Steady State Insertion Limit~~s~~ and the Power Dependent Insertion Limit~~s~~ for intervals ~~greater than~~ 5 EFPD per 30 EFPD interval or ~~greater than~~ 14 EFPD per ~~calendar year, either:~~

Required Action C.1

1. Restore ~~the~~ regulating groups to within ~~the Long Term Steady State Insertion~~ Limits within 2 hours~~, or~~

Required Action E.1

2. Be in ~~at least HOT STANDBY~~ within 6 hours.

SR 3.1.6.1 NOTE

NOTE Not required to be performed until 12 hours after entry into MODE 2. L02

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

~~4.1.3.6~~ The position of each regulating CEA group ~~shall be determined to be~~ within ~~the Power Dependent~~ Insertion Limits in accordance with the Surveillance Frequency Control Program ~~except during time intervals when the~~ PDIL ~~Auctioneer~~ Alarm Circuit ~~is~~ inoperable, ~~then verify the individual CEA positions at least~~ once per 4 hours.

ACTION D
Required Action D.1

SR 3.1.6.2

The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Power Dependent Insertion Limits ~~shall be determined~~ in accordance with the Surveillance Frequency Control Program.

1 hour AND M02

Perform SR 3.1.6.1.

SR 3.1.6.3

Verify PDIL alarm circuit is OPERABLE in accordance with the Surveillance Frequency Control Program. A04

A01

ITS

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

Applicability **APPLICABILITY:** MODES 1, 2*, 3 and 4.

See ITS 3.1.1

ACTION:

With the SHUTDOWN MARGIN outside the COLR limits, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

See ITS 3.1.4

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

SR 3.1.6.1

b. ~~When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0,~~ in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

L03

c. ~~When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.~~

~~* See Special Test Exception 3.10.1.~~

A02

3/4.2 POWER DISTRIBUTION LIMITS

See ITS 3.2.1

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent control limits of COLR Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within 1 hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.3 Excore Detector Monitoring System – The excore detector monitoring system may be used for monitoring the linear heat rate by:

SR 3.1.6.1
SR 3.1.6.2

a. Verifying in accordance with the Surveillance Frequency Control Program that the ~~full-length~~ CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.

A05

b. Verifying in accordance with the Surveillance Frequency Control Program that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on COLR Figure 3.2-2.

DISCUSSION OF CHANGES
ITS 3.1.6, REGULATING CONTROL ELEMENT ASSEMBLY (CEA) INSERTION LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.1.3.6 Applicability is MODES 1 and 2 with a footnote (footnote *) for MODE 2 stating "See Special Test Exception 3.10.2 and 3.10.5," and a footnote (footnote #) for MODE 2 stating "With $K_{eff} \geq 1.0$." CTS 3.1.1.1 Applicability is, in part, MODE 2 with a footnote (footnote *) stating "See Special Test Exception 3.10.1," and Unit 1 CTS 4.1.1.1.1.b and c contain footnotes (footnotes # and ##) for MODE 2 stating "With $K_{eff} \geq 1.0$," and "With $K_{eff} < 1.0$," respectively. ITS 3.1.6 does not contain the footnotes or a reference to the Special Test Exception or reference to the K_{eff}. This changes the CTS by not including footnote * and footnotes # and ## in the ITS.

The purpose of the footnote references is to alert the user that conditions exist that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS. See DOC M01 for additional discussion of change for the footnote # for MODE 2 stating "With $K_{eff} \geq 1.0$.''

A03 **Unit 1 only:** PSL Unit 1 CTS 3.1.3.6 ACTION b. states, in part, that power operation may proceed provided ACTIONS b.1or b.2 are met, "except during operation pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1." PSL Unit 1 CTS 3.1.3.6 ACTION c. states, in part, that power operation may proceed provided ACTIONS c.1 or c.2 are met, "except during operation pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1." ITS 3.1.6 does not provide a reference to ITS 3.1.4.

The purpose of the CTS statement "except during operation pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1." is to provide a reference to another Specification that may apply under these conditions. This changes the CTS by not including the cross-reference in the ITS. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

A04 CTS 4.1.3.6 states that with the Power Dependent Insertion Limits (PDIL) alarm circuit inoperable, verification of the individual CEA positions is required once per 4 hours. However, the PDIL alarm circuit operability is not specifically addressed. CTS does not require that the PDIL alarm circuit be OPERABLE in the LCO statement, and CTS does not provide a Surveillance to verify the PDIL

alarm circuit is OPERABLE. ITS 3.1.6 LCO requires the PDIL alarm circuit be OPERABLE, and ITS SR 3.1.6.3 verifies the PDIL alarm circuit is OPERABLE. | R1

The purpose of CTS 4.1.3.6 is to ensure that with the Power Dependent Insertion Limits (PDIL) alarm circuit is OPERABLE, since CTS requires ACTIONS be taken if the PDIL alarm circuit is inoperable. To ensure the position of the individual CEA positions is within limits to avoid improper CEA alignment, it is necessary to initially verify the individual CEA positions within one hour. These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A05 PSL Unit 1 CTS 4.2.1.3.a and PSL Unit 2 CTS 4.2.1.3.a include the descriptor "full length" in reference to CEA(s). The purpose of the CTS 3.1.3.1 descriptor is to distinguish between full length and partial length CEA(s). PSL Unit 1 and PSL Unit 2 no longer use partial length CEA(s). Therefore, since partial length CEA(s) are no longer a design feature for PSL Unit 1 and PSL Unit 2, the "full length" descriptors used for CEA(s) are deleted.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.1.3.6 is applicable in MODES 1 and 2 with $k_{eff} \geq 1.0$. MODE 2 is modified by CTS 3.1.3.6 footnote #. ITS 3.1.6 is applicable in MODES 1 and 2. This changes the CTS by expanding the Applicability from MODE 2 with the reactor critical to all of MODE 2.

The purpose of CTS 3.1.3.6 is to maintain the regulating CEA sequence, overlap, and physical insertion limits with the reactor in MODES 1 and 2 with the reactor critical. These limits must be maintained, since they preserve power distribution, ejected CEA worth, SDM, and reactivity rate insertion assumptions. This change is acceptable because applying the requirement prior to bringing the reactor critical ensures The limits on regulating CEAs sequence, overlap, and physical insertion, as defined in the COLR, are maintained and serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected CEA worth is maintained, and ensuring adequate negative reactivity insertion on trip. This change is designated as more restrictive because it increases the conditions under which Technical Specification controls will be applied.

M02 CTS 4.1.3.6 states that with the Power Dependent Insertion Limits (PDIL) alarm circuit inoperable, verification of the individual CEA positions is required once per 4 hours. ITS 3.1.6 ACTION D requires with the PDIL alarm circuit inoperable, verification of the individual CEA positions must be completed within 1 hour and once per 4 hours thereafter. This changes the CTS by requiring the initial verification of the individual CEA positions be performed within 1 hour.

The purpose of CTS 4.1.3.6 is to ensure that with the Power Dependent Insertion Limits (PDIL) alarm circuit inoperable, improper CEA alignments are identified before unacceptable flux distributions occur. To ensure the position of the individual CEA positions is within limits to avoid improper CEA alignment, it is necessary to initially verify the individual CEA positions within one hour. This change is designated as more restrictive because it increases the conditions under which Technical Specification controls will be applied.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (Type 1 – Removing Details of System Design and System Description, Including Design Limits) CTS 3.1.3.6 LCO states that regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches. ITS LCO 3.1.6 does not retain this detail. This changes the CTS by relocating the details that regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches to the Bases.

The removal of these details, that are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for the regulating CEA groups to be limited to the withdrawal sequence and to the insertion limits specified in the COLR) with CEA insertion between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.1.3.6 ACTION b. states that with the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits for intervals > 4 hours per 24 hour interval, operation may proceed provided either (CTS ACTION b.1) the Short Term Steady State Insertion Limits are not exceeded, or (CTS ACTION b.2) any subsequent increase in THERMAL POWER is restricted to < 5% of RATED THERMAL POWER per hour. The CTS completion time is assumed to be “immediately” since a Completion Time for ACTION b. is not provided. ITS 3.1.6, Required Action B.1 requires verification within 15 minutes

that the Short Term Steady State Insertion Limits are not exceeded. ITS 3.1.6 Required Action B.2 requires that within 15 minutes any subsequent increase in THERMAL POWER is restricted to < 5% of RATED THERMAL POWER per hour. This changes the CTS by relaxing the Completion Time from "immediately" to 15 minutes.

The purpose of CTS 3.1.2.6 ACTION b. is to ensure peaking factors remain within limits with the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits for extended intervals. If the CEAs are inserted between the long term steady state insertion limits and the transient insertion limits for intervals > 4 hours per 24 hour period, and the short term steady state insertions are exceeded, peaking factors can develop that are of immediate concern. Fifteen minutes provides adequate time for the operator to verify if the short term steady state insertion limits are exceeded. Experience has shown that rapid power increases in areas of the core, in which the flux has been depressed, can result in fuel damage, as the LHR in those areas rapidly increases. Restricting the rate of THERMAL POWER increases to ≤ 5% RTP per hour, following CEA insertion beyond the long term steady state insertion limits, ensures the power transients experienced by the fuel will not result in fuel failure

This change is acceptable because the Completion Time is consistent with safe operation under the specific Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, and the low probability of a DBA occurring during the allowed Completion Time. This ITS Completion Time of 15 minutes is adequate for an operator to perform the verification and implement measures to restrict the rate of THERMAL POWER increases.

This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L02 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.1.3.6 requires verification of regulating CEA groups position and regulating CEA groups time Inside Long Term Steady State Insertion Limits in accordance with the Surveillance Frequency Control Program (SFCP). The SFCP frequencies are at least once per 12 hours and at least per 24 hours, respectively. ITS SR 3.1.6.1 provides a Note that states “Not required to be performed until 12 hours after entry into MODE 2” indicating that entry is allowed into MODE 2 for 12 hours without having performed the SR. This is necessary, since the unit must be in the applicable MODES in order to perform Surveillances that demonstrate the LCO limits are met. CTS does not contain this Note. CTS SR 4.0.4 and CTS LCO 3.0.4 allow entry into 3.1.3.6 ACTION a to allow 2 hours to complete the Surveillance in MODE 2. | R1

The purpose of CTS 4.1.3.6 is to verify regulating CEA groups position and regulating CEA groups time Inside Long Term Steady State Insertion Limits in accordance with the Surveillance Frequency Control Program (SFCP). ITS 3.1.6 Note indicates that entry is allowed into MODE 2 for 12 hours without having performed the SR. This is necessary, since the unit must be in MODE 2 in order

to perform the Surveillance that verifies each regulating CEA group position is within its insertion limits. CTS does not contain this Note. CTS SR 4.0.4 and CTS LCO 3.0.4 allow entry into 3.1.3.6 ACTION a to allow 2 hours to complete the Surveillance in MODE 2. | R1

This change is designated as less restrictive because additional time is allowed to perform the Surveillances in MODE 2 in the ITS that was not allowed in the CTS. | R1

L03 (*Category 5 – Deletion of Surveillance Requirement*) CTS 4.1.1.1.1.c requires verification of SDM, when in MODE 2 with keff < 1.0, at least once during CEA withdrawal and in accordance with the Surveillance Frequency Control Program until the reactor is critical. ITS 3.1.6 does not require an explicit requirement to verify SDM in MODE 2 with keff < 1.0. This changes the CTS by eliminating the explicit statement that a SDM verification is required to be performed in MODE 2 with keff < 1.0.

The purpose of CTS 4.1.1.1.1.c is to verify SDM is within the required limit more frequently from entry into MODE 2 until the reactor is critical. Limits on shutdown and regulating CEA insertion ensure SDM limits are preserved. This change is acceptable because the shutdown CEAs must be withdrawn to ≥ 129 inches and the regulating CEAs are required to be within the insertion limits specified in the COLR upon entry into MODE 2 per the requirements of CTS 3.0.1 (ITS LCO 3.0.1) and CTS 3.0.4 (ITS LCO 3.0.4). Therefore, it is unnecessary to verify SDM more frequently in MODE 2 until the reactor is critical. ITS SR 3.1.1.1 requires SDM to verified in accordance with the Surveillance Frequency Control Program while in MODE 3 and, prior to entry into MODE 2, SR 3.1.5.1 and 3.1.6.1 (i.e., verifying shutdown and regulating CEAs are within the required insertion limits) must be met per ITS SR 3.0.4. Thus, there is continued confidence that SDM will be within limits during a plant startup in MODE 2 until the reactor is critical. The surveillance testing associated with SDM and CEA insertion limits is considered adequate to assure, pursuant to the requirements of 10 CFR 50.36(c)(3), that facility operation will be within safety limits and that the limiting condition for operation associated with the shutdown and regulating CEA insertion limits will be met. This change is designated as less restrictive because a Surveillance that was required in the CTS will not be performed in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~ 3

3.1.6 Regulating Control Element Assembly (CEA) Insertion Limits ~~(Analog)~~

LCO 3.1.3.6

LCO 3.1.6 The power dependent insertion limit (PDIL) alarm circuit shall be OPERABLE, and the regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits specified in the COLR.

Applicability

APPLICABILITY: MODES 1 and 2.

Action a.

----------NOTE----------
This LCO is not applicable while performing SR 3.1.4.4 ~~[or during reactor power cutback operation]~~. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a., a.1 A. Regulating CEA groups inserted beyond the ~~transient~~ insertion limit.	A.1 Restore regulating CEA groups to within limits. 1 OR Action a.2 A.2 Reduce THERMAL POWER to less than or equal to the fraction of RTP allowed by the CEA group position and insertion limits specified in the COLR.	2 hours 2 hours
Action b., b.1 B. Regulating CEA groups inserted between the long term steady state insertion limit and the ~~transient~~ insertion limit for > 4 hours per 24 hour interval.	B.1 Verify short term steady state insertion limits are not exceeded. 1 OR Action b.2 B.2 Restrict increases in THERMAL POWER to ≤ 5% RTP per hour.	15 minutes 15 minutes

power dependent

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
power dependent Action C, c.1 C. Regulating CEA groups inserted between the long term steady state insertion limit and the ~~transient~~ insertion limit for intervals > 5 effective full power days (EFPD) per 30 EFPD interval or > 14 EFPD per 365 EFPD.	C.1 Restore regulating CEA groups to within limits.	2 hours
D. PDIL alarm circuit inoperable.	D.1 Perform SR 3.1.6.1.	1 hour AND Once per 4 hours thereafter
Action c.2 E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.6.1	----------NOTE---------- Not required to be performed until 12 hours after entry into MODE 2. ----------	
SR 4.1.3.6 SR 4.1.1.1.1.b	Verify each regulating CEA group position is within its insertion limits.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2
SR 4.1.3.6 SR 3.1.6.2	Verify the accumulated times during which the regulating CEA groups are inserted beyond the steady state insertion limits but within the ~~transient~~ power dependent insertion limits.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2
SR 4.1.3.6 SR 3.1.6.3	Verify PDIL alarm circuit is OPERABLE.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

3

3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

3

3.1.6 Regulating Control Element Assembly (CEA) Insertion Limits ~~(Analog)~~

LCO 3.1.3.6

LCO 3.1.6 The power dependent insertion limit (PDIL) alarm circuit shall be OPERABLE, and the regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits specified in the COLR.

Applicability

APPLICABILITY: MODES 1 and 2.

Action a.

----------NOTE----------
This LCO is not applicable while performing SR 3.1.4.4 ~~[or during reactor power cutback operation]~~.

2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a., a.1 A. Regulating CEA groups inserted beyond the ~~transient~~ insertion limit.	A.1 Restore regulating CEA groups to within limits. OR Action a.2 A.2 Reduce THERMAL POWER to less than or equal to the fraction of RTP allowed by the CEA group position and insertion limits specified in the COLR.	2 hours 2 hours
Action b., b.1 B. Regulating CEA groups inserted between the long term steady state insertion limit and the ~~transient~~ insertion limit for > 4 hours per 24 hour interval.	B.1 Verify short term steady state insertion limits are not exceeded. OR Action b.2 B.2 Restrict increases in THERMAL POWER to ≤ 5% RTP per hour.	15 minutes 15 minutes

power dependent

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
power dependent Action C, c.1 C. Regulating CEA groups inserted between the long term steady state insertion limit and the ~~transient~~ insertion limit for intervals > 5 effective full power days (EFPD) per 30 EFPD interval or > 14 EFPD per 365 EFPD.	C.1 Restore regulating CEA groups to within limits.	2 hours
D. PDIL alarm circuit inoperable.	D.1 Perform SR 3.1.6.1.	1 hour AND Once per 4 hours thereafter
Action c.2 E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.6.1	----------NOTE---------- Not required to be performed until 12 hours after entry into MODE 2. ----------	
SR 4.1.3.6 SR 4.1.1.1.1.b	Verify each regulating CEA group position is within its insertion limits.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2
SR 4.1.3.6 SR 3.1.6.2	Verify the accumulated times during which the regulating CEA groups are inserted beyond the steady state insertion limits but within the ~~transient~~ power dependent insertion limits.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 2
SR 4.1.3.6 SR 3.1.6.3	Verify PDIL alarm circuit is OPERABLE.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.6, REGULATING CONTROL ELEMENT ASSEMBLY (CEA) INSERTION LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

6

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

6

B 3.1.6 Regulating Control Element Assembly (CEA) Insertion Limits ~~(Analog)~~

BASES

BACKGROUND

The insertion limits of the regulating Control Element Assemblies (CEAs) are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected CEA worth, reactivity insertion rate, and SDM limits are preserved.

The regulating CEA groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between CEA worth and CEA position (integral CEA worth). The regulating CEA groups are withdrawn and operate in a predetermined sequence. The group sequence and overlap limits are specified in the COLR.

The regulating CEAs are used for precise reactivity control of the reactor. The positions of the regulating CEAs are manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.6, LCO 3.2.~~4~~, "AZIMUTHAL POWER TILT (Tq)," and LCO 3.2.~~5~~, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LCO 3.2.1, "Linear Heat Rate (LHR)"), ~~total planar radial peaking factor (F_{XY}^T) (LCO 3.2.2, "Total Planar Radial Peaking Factor (F_{XY}^T)"),~~ and total integrated radial peaking factor (F_r^T) (LCO 3.2.3, "Total Integrated Radial Peaking Factor (F_r^T)") limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling System analysis. Operation within the ~~F_{XY}^T and~~

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BASES

BACKGROUND (continued)

F_r^T limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the LHR~~, F_{XY}^T,~~ and F^T_r limits, certain reactivity limits are preserved by regulating CEA insertion limits. The regulating CEA insertion limits also restrict the ejected CEA worth to the values assumed in the safety analysis and preserve the minimum required SDM in MODES 1 and 2.

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The establishment of limiting safety system settings and LCOs requires that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks, due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). From these analyses, CEA insertions are determined and a consistent set of radial peaking factors defined. The long term steady state and short term insertion limits are determined, based upon the assumed mode of operation used in the analyses, and provide a means of preserving the assumption on CEA insertions used. The long and short term insertion limits of LCO 3.1.6 are specified for the plant, which has been designed primarily for base loaded operation, but has the ability to accommodate a limited amount of load maneuvering.

The regulating CEA insertion and alignment limits are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the regulating bank insertion limits control the reactivity that could be added in the event of a CEA ejection accident, and the shutdown and regulating bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow, ejected CEA, or other accident requiring termination by a Reactor Protection System trip function.

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BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The acceptance criteria for the regulating CEA insertion, ASI, and T_q LCOs are such as to preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F, 10 CFR 50.46 (Ref. 2),

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel CEA in the core does not experience a DNB condition,

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 2~~8~~0 cal/gm (Ref. 3), and

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The SDM requirement is ensured by limiting the regulating and shutdown CEA insertion limits, so that the allowable inserted worth of the CEAs is such that sufficient reactivity is available to shut down the reactor to hot zero power. SDM assumes the maximum worth CEA remains fully withdrawn upon trip (Ref. 4).

The most limiting SDM requirements for MODE 1 and 2 conditions at BOC are determined by the requirements of several transients, e.g., loss of flow, seized rotor, etc. However, the most limiting SDM requirements for MODES 1 and 2 at EOC come from just one transient, Steam Line Break (SLB). The requirements of the SLB event at EOC for both the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Although the most limiting SDM requirements at EOC are much larger than those at BOC, the available SDMs obtained via the scramming of the CEAs are also substantially larger due to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle are sufficient since the differences between available SDMs and the limiting SDM requirements are the smallest at these times in cycle. The measurement of CEA bank worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adher~~a~~nce to LCOs 3.1.5 and 3.1.6 provides assurance that the available SDMs at any time in cycle will exceed the limiting SDM requirements at that time in cycle.

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Operation at the insertion limits or ASI limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed T_q present. Operation at the insertion limit may also indicate the maximum ejected CEA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected CEA worths.

The regulating and shutdown CEA insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected CEA worth, and power distribution peaking factors are preserved (Ref. ~~5~~).

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The regulating CEA insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on regulating CEAs sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected CEA worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating CEA motion.

Regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches

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The power dependent insertion limit (PDIL) alarm circuit is required to be OPERABLE for notification that the CEAs are outside the required insertion limits. When the PDIL alarm circuit is inoperable, the verification of CEA positions is increased to ensure improper CEA alignment is identified before unacceptable flux distribution occurs.

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BASES

APPLICABILITY

The regulating CEA sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected CEA worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected CEA worth assumptions would be exceeded in these MODES. SDM is preserved in MODES 3, 4, and 5 by adjustments to the soluble boron concentration.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.4. This SR verifies the freedom of the CEAs to move, and requires the regulating CEAs to move below the LCO limits, which would normally violate the LCO. ~~The Note also allows the LCO to be not applicable during reactor power cutback operation, which inserts a selected CEA group (usually group 5) during loss of load events.~~

5

ACTIONS

A.1 and A.2

Operation beyond the transient insertion limit may result in a loss of SDM and excessive peaking factors. The transient insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the CEAs in response to changing plant conditions. When the regulating groups are inserted beyond the transient insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual CEA insertion limit. Two hours provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

B.1 and B.2

If the CEAs are inserted between the long term steady state insertion limits and the transient insertion limits for intervals > 4 hours per 24 hour period, and the short term steady state insertions are exceeded, peaking factors can develop that are of immediate concern (Ref. ~~6~~ 3).

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Verifying the short term steady state insertion limits are not exceeded ensures that the peaking factors that do develop are within those allowed for continued operation. Fifteen minutes provides adequate time for the operator to verify if the short term steady state insertion limits are exceeded.

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BASES

ACTIONS (continued)

Experience has shown that rapid power increases in areas of the core, in which the flux has been depressed, can result in fuel damage, as the LHR in those areas rapidly increases. Restricting the rate of THERMAL POWER increases to ≤ 5% RTP per hour, following CEA insertion beyond the long term steady state insertion limits, ensures the power transients experienced by the fuel will not result in fuel failure (Ref. ~~7~~).

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C.1

With the regulating CEAs inserted between the long term steady state insertion limit and the ~~transient~~ insertion limit, and with the core approaching the 5 effective full power days (EFPD) per 30 EFPD or 14 EFPD per 365 EFPD limits, the core approaches the acceptable limits placed on operation with flux patterns outside those assumed in the long term burnup assumptions (Ref. ~~8~~). In this case, the CEAs must be returned to within the long term steady state insertion limits, or the core must be placed in a condition in which the abnormal fuel burnup cannot continue. A Completion Time of 2 hours is allotted to return the CEAs to within the long term steady state insertion limits.

power dependent

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The required Completion Time of 2 hours from initial discovery of a regulating CEA group outside the limits until its restoration to within the long term steady state limits, shown on the figures in the COLR, allows sufficient time for borated water to enter the Reactor Coolant System from the chemical addition and makeup systems, and ~~to cause~~ the regulating CEAs to ~~withdraw~~ to the acceptable region. It is reasonable to continue operation for 2 hours after it is discovered that the 5 day or 14 day EFPD limit has been exceeded. This Completion Time is based on limiting the potential xenon redistribution, the low probability of an accident, and the steps required to complete the action.

allowing

be withdrawn

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D.1

When the PDIL alarm circuit is inoperable, performing SR 3.1.6.1 within 1 hour and once per 4 hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur.

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BASES

ACTIONS (continued)

E.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

With the PDIL alarm circuit OPERABLE, verification of each regulating CEA group position is sufficient to detect CEA positions that may approach the acceptable limits, and to provide the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded. ~~[The 12 hour Frequency also takes into account the indication provided by the PDIL alarm circuit and other information about CEA group positions available to the operator in the control room.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.1.6.1 is modified by a Note indicating that entry is allowed into MODE 2 for 12 hours without having performed the SR. This is necessary, since the unit must be in the applicable MODES in order to perform Surveillances that demonstrate the LCO limits are met.

SR 3.1.6.2

Verification of the accumulated time of CEA group insertion between the long term steady state insertion limits and the ~~transient~~ insertion limits ensures the cumulative time limits are not exceeded. ~~[The 24 hour Frequency ensures the operator identifies a time limit that is being approached before it is reached.~~

power dependent

2

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BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.1.6.3

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. ~~[The 31 day Frequency takes into account other Surveillances being performed at shorter Frequencies that identify improper CEA alignments.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.

2. 10 CFR 50.46

3. FSAR, Section ~~[ ], Section [ ], and Section [ ]~~.
U 15

~~4. FSAR, Section [ ].~~

~~5. FSAR, Section [ ].~~

~~6. FSAR, Section [ ].~~

2

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BASES

REFERENCES (continued)

~~7. FSAR, Section [ ].~~

~~8. FSAR, Section [ ].~~

2

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B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

6

B 3.1.6 Regulating Control Element Assembly (CEA) Insertion Limits ~~(Analog)~~

BASES

BACKGROUND

The insertion limits of the regulating Control Element Assemblies (CEAs) are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected CEA worth, reactivity insertion rate, and SDM limits are preserved.

The regulating CEA groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between CEA worth and CEA position (integral CEA worth). The regulating CEA groups are withdrawn and operate in a predetermined sequence. The group sequence and overlap limits are specified in the COLR.

The regulating CEAs are used for precise reactivity control of the reactor. The positions of the regulating CEAs are manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.6, LCO 3.2.~~4~~, "AZIMUTHAL POWER TILT (Tq)," and LCO 3.2.~~5~~, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LCO 3.2.1, "Linear Heat Rate (LHR)"), ~~total planar radial peaking factor (F^T_{XY}) (LCO 3.2.2, "Total Planar Radial Peaking Factor (F^T_{XY})"),~~ and total integrated radial peaking factor (F^T_r) (LCO 3.2.3, "Total Integrated Radial Peaking Factor (F^T_r)") limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling System analysis. Operation within the ~~F^T_{XY} and~~

3 4 3 3 3

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BASES

BACKGROUND (continued)

F_r^T limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the LHR~~, F_{XY}^T,~~ and F^T_r limits, certain reactivity limits are preserved by regulating CEA insertion limits. The regulating CEA insertion limits also restrict the ejected CEA worth to the values assumed in the safety analysis and preserve the minimum required SDM in MODES 1 and 2.

3

The establishment of limiting safety system settings and LCOs requires that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks, due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). From these analyses, CEA insertions are determined and a consistent set of radial peaking factors defined. The long term steady state and short term insertion limits are determined, based upon the assumed mode of operation used in the analyses, and provide a means of preserving the assumption on CEA insertions used. The long and short term insertion limits of LCO 3.1.6 are specified for the plant, which has been designed primarily for base loaded operation, but has the ability to accommodate a limited amount of load maneuvering.

The regulating CEA insertion and alignment limits are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the regulating bank insertion limits control the reactivity that could be added in the event of a CEA ejection accident, and the shutdown and regulating bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow, ejected CEA, or other accident requiring termination by a Reactor Protection System trip function.

1

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The acceptance criteria for the regulating CEA insertion, ASI, and T_q LCOs are such as to preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F, 10 CFR 50.46 (Ref. 2),

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel CEA in the core does not experience a DNB condition,

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 2~~8~~0 cal/gm (Ref. 3), and

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The SDM requirement is ensured by limiting the regulating and shutdown CEA insertion limits, so that the allowable inserted worth of the CEAs is such that sufficient reactivity is available to shut down the reactor to hot zero power. SDM assumes the maximum worth CEA remains fully withdrawn upon trip (Ref. ~~4~~3).

The most limiting SDM requirements for MODE 1 and 2 conditions at BOC are determined by the requirements of several transients, e.g., loss of flow, seized rotor, etc. However, the most limiting SDM requirements for MODES 1 and 2 at EOC come from just one transient, Steam Line Break (SLB). The requirements of the SLB event at EOC for both the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

6

BASES

APPLICABLE SAFETY ANALYSES (continued)

Although the most limiting SDM requirements at EOC are much larger than those at BOC, the available SDMs obtained via the scramming of the CEAs are also substantially larger due to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle are sufficient since the differences between available SDMs and the limiting SDM requirements are the smallest at these times in cycle. The measurement of CEA bank worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adher~~a~~nce to LCOs 3.1.5 and 3.1.6 provides assurance that the available SDMs at any time in cycle will exceed the limiting SDM requirements at that time in cycle.

e

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Operation at the insertion limits or ASI limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed T_q present. Operation at the insertion limit may also indicate the maximum ejected CEA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected CEA worths.

The regulating and shutdown CEA insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected CEA worth, and power distribution peaking factors are preserved (Ref. ~~5~~).

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The regulating CEA insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on regulating CEAs sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected CEA worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating CEA motion.

Regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches

4

The power dependent insertion limit (PDIL) alarm circuit is required to be OPERABLE for notification that the CEAs are outside the required insertion limits. When the PDIL alarm circuit is inoperable, the verification of CEA positions is increased to ensure improper CEA alignment is identified before unacceptable flux distribution occurs.

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6

BASES

APPLICABILITY

The regulating CEA sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected CEA worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected CEA worth assumptions would be exceeded in these MODES. SDM is preserved in MODES 3, 4, and 5 by adjustments to the soluble boron concentration.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.4. This SR verifies the freedom of the CEAs to move, and requires the regulating CEAs to move below the LCO limits, which would normally violate the LCO. ~~The Note also allows the LCO to be not applicable during reactor power cutback operation, which inserts a selected CEA group (usually group 5) during loss of load events.~~

5

ACTIONS

A.1 and A.2

Operation beyond the transient insertion limit may result in a loss of SDM and excessive peaking factors. The transient insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the CEAs in response to changing plant conditions. When the regulating groups are inserted beyond the transient insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual CEA insertion limit. Two hours provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

B.1 and B.2

If the CEAs are inserted between the long term steady state insertion limits and the transient insertion limits for intervals > 4 hours per 24 hour period, and the short term steady state insertions are exceeded, peaking factors can develop that are of immediate concern (Ref. ~~6~~ 3).

1

Verifying the short term steady state insertion limits are not exceeded ensures that the peaking factors that do develop are within those allowed for continued operation. Fifteen minutes provides adequate time for the operator to verify if the short term steady state insertion limits are exceeded.

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BASES

ACTIONS (continued)

Experience has shown that rapid power increases in areas of the core, in which the flux has been depressed, can result in fuel damage, as the LHR in those areas rapidly increases. Restricting the rate of THERMAL POWER increases to ≤ 5% RTP per hour, following CEA insertion beyond the long term steady state insertion limits, ensures the power transients experienced by the fuel will not result in fuel failure (Ref. ~~7~~).

3

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C.1

With the regulating CEAs inserted between the long term steady state insertion limit and the ~~transient~~ insertion limit, and with the core approaching the 5 effective full power days (EFPD) per 30 EFPD or 14 EFPD per 365 EFPD limits, the core approaches the acceptable limits placed on operation with flux patterns outside those assumed in the long term burnup assumptions (Ref. ~~8~~). In this case, the CEAs must be returned to within the long term steady state insertion limits, or the core must be placed in a condition in which the abnormal fuel burnup cannot continue. A Completion Time of 2 hours is allotted to return the CEAs to within the long term steady state insertion limits.

power dependent

3

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The required Completion Time of 2 hours from initial discovery of a regulating CEA group outside the limits until its restoration to within the long term steady state limits, shown on the figures in the COLR, allows sufficient time for borated water to enter the Reactor Coolant System from the chemical addition and makeup systems, and ~~to cause~~ the regulating CEAs to ~~withdraw~~ to the acceptable region. It is reasonable to continue operation for 2 hours after it is discovered that the 5 day or 14 day EFPD limit has been exceeded. This Completion Time is based on limiting the potential xenon redistribution, the low probability of an accident, and the steps required to complete the action.

allowing

be withdrawn

1

D.1

When the PDIL alarm circuit is inoperable, performing SR 3.1.6.1 within 1 hour and once per 4 hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur.

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Revision XXX

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6

BASES

ACTIONS (continued)

E.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

With the PDIL alarm circuit OPERABLE, verification of each regulating CEA group position is sufficient to detect CEA positions that may approach the acceptable limits, and to provide the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded. ~~[The 12 hour Frequency also takes into account the indication provided by the PDIL alarm circuit and other information about CEA group positions available to the operator in the control room.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.1.6.1 is modified by a Note indicating that entry is allowed into MODE 2 for 12 hours without having performed the SR. This is necessary, since the unit must be in the applicable MODES in order to perform Surveillances that demonstrate the LCO limits are met.

SR 3.1.6.2

Verification of the accumulated time of CEA group insertion between the long term steady state insertion limits and the ~~transient~~ insertion limits ensures the cumulative time limits are not exceeded. ~~[The 24 hour Frequency ensures the operator identifies a time limit that is being approached before it is reached.~~

power dependent

2

1

6

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.1.6.3

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. ~~[The 31 day Frequency takes into account other Surveillances being performed at shorter Frequencies that identify improper CEA alignments.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.

2. 10 CFR 50.46

3. FSAR, Section ~~[ ]~~, Section ~~[ ]~~, and Section ~~[ ]~~.
   U 15

~~4. FSAR, Section [ ].~~

~~5. FSAR, Section [ ].~~

~~6. FSAR, Section [ ].~~

2

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6

BASES

REFERENCES (continued)

~~7. FSAR, Section [ ].~~

~~8. FSAR, Section [ ].~~

2

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.6, BASES, REGULATING CONTROL ELEMENT ASSEMBLY (CEA) INSERTION LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS includes ISTS 3.2.1, Linear Heat Rate (LHR), ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}), ISTS 3.2.3, Total Integrated Radial Peaking Factor (F_{r}^{T}), ISTS 3.2.4, Azimuthal Power Tilt (T_q), and ISTS 3.2.5, Axial Shape Index (ASI). CTS do not include a Specification for ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}). The CTS and ISTS are renumbered. CTS 3.2.3, Total Integrated Radial Peaking Factor (F_{r}^{T}), CTS 3.2.4, Azimuthal Power Tilt (T_q), and CTS 3.2.5, Axial Shape Index (ASI), are renumbered as ITS 3.2.2, ITS 3.2.3, and ITS 3.2.4, respectively.

4. CTS 3.1.3.6 LCO states that regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches. ITS LCO 3.1.6 does not retain this detail. This changes the CTS by relocating the details that regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches to the Bases. See DOC LA01.

5. The ISTS Bases states “The Note also allows the LCO to be not applicable during reactor power cutback operation, which inserts a selected CEA group (usually group 5) during loss of load events.” Reactor power cutback design feature is not applicable to PSL Unit 1 and Unit 2. Therefore, this statement is deleted.

6. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.6, REGULATING CONTROL ELEMENT ASSEMBLY (CEA) INSERTION LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

3.1.7 Control Element Assembly (CEA) Position Indication

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

LCO 3.1.7

~~3.1.3.3~~ ~~All shutdown and~~ regulating CEA reed switch position indicator channel~~s~~ and ~~CEA~~ pulse counting position indicator channel~~s~~ shall be OPERABLE ~~and capable of determining the absolute CEA positions within ± 2.25 inches.~~

for each CEA

A05

Applicability

APPLICABILITY: MODES 1 and 2.

ACTION:

Separate Condition entry is allowed for each CEA group.

L01

~~a. Deleted.~~

One or more CEA groups with

inoperable

Condition A

~~b.~~ ~~With a maximum of~~ one reed switch position indicator channel ~~per group~~ or one ~~(except as permitted by ACTION item d. below)~~ pulse counting position indicator channel ~~per group~~ inoperable ~~and the CEA(s) with the~~ inoperable position indicator channel

A02

Condition A Note

is associated with

Only applicable when partially inserted, within 6 hours either:

ITS Required Action A.1, A.2, and Note

L04

CEA

~~1. Restore the inoperable position indicator channel to OPERABLE status, or~~

MODE 3

Required Action D.1

~~2.~~ Be in ~~HOT STANDBY, or~~

RTP

Required Action A.3.1

~~3.~~ Reduce THERMAL POWER to ≤ 70% ~~of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used.~~ ~~Operation at or below this reduced THERMAL POWER level may continue provided that~~ within the next 4 hours ~~either:~~

A03

A04

Required Action A.3.2.1

~~a)~~ ~~The~~ CEA ~~group(s)~~ with ~~the~~ inoperable position indicator ~~is~~ fully withdrawn ~~while~~ maintaining the withdrawal sequence required by Specification 3.1.3.6 ~~and when this~~ CEA ~~group reaches its fully withdrawn position, the~~ "Full Out" limit ~~of the CEA with the inoperable~~ position indicator is actuated ~~and verifies this CEA to be fully withdrawn.~~ ~~Subsequent to fully withdrawing this CEA group(s), the THERMAL POWER level may be returned to a level consistent with all other applicable specifications;~~ or

within 10 hours

Required Action A.3.2.1 Note

A01

Verify associated

Required Action B.1

A04

A01

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS (Continued)

LIMITING CONDITION FOR OPERATION

within 6 hours [A.2], within 10 hours [A.3.2.2]

Required Action A.3.2.2
Required Action A.3.2.2 Note
Required Action B.2
Verify associated

~~b)~~ ~~The~~ CEA ~~group(s)~~ with the inoperable position indicator ~~is~~ fully inserted~~, and subsequently maintained fully inserted,~~ while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 ~~and when this~~ CEA ~~group reaches its fully inserted position, the~~ "Full In" limit ~~of the CEA with the inoperable~~ position indicator is actuated ~~and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6~~.

A01

A04

One or more CEA groups with
inoperable
Condition B

~~c.~~ ~~With a maximum of~~ one reed switch position indicator channel ~~per group~~ or one pulse counting position indicator channel ~~per group~~ inoperable ~~and the CEA(s) with the~~ inoperable position indicator channel ~~at either its~~ fully inserted ~~position~~ or ~~fully~~ withdrawn ~~position , operation may continue provided:~~

Condition B Note
is associated with
Only applicable when
CEA

Required Action B.1 and B.2
Verify associated

1. ~~The position of this~~ CEA ~~is verified~~ immediately and ~~at least~~ once per 12 hours thereafter ~~by its~~ "Full In" or "Full Out" limit ~~(as applicable),~~

position indicator is actuated

2. ~~The fully inserted CEA group(s) containing the inoperable position indicator channel is subsequently maintained fully inserted, and~~

A06

3. ~~Subsequent operation is within the limits of Specification 3.1.3.6.~~

A04

One or more groups
Condition C

~~d.~~ With ~~one~~ or more pulse counting position indicator channels
two
inoperable~~, operation in MODES 1 and 2 may continue for up to 24~~ hours ~~provided all of the~~ reed switch position indicator channel~~s are OPERABLE~~.

L02

Verify no more than one per group is inoperable

Require Action C.1

L03

Require Action C.2
Restore pulse counting position indicator channels to OPERABLE status within 72

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

~~4.1.3.3~~ ~~Each position indicator channel shall be determined to be OPERABLE by~~ verify~~ing the~~ pulse counting position indicator channels and ~~the~~ reed switch position indicator channels agree within 4.5 inches in accordance with the Surveillance Frequency Control Program ~~except during time intervals~~ when ~~the~~ Deviation circuit is inoperable~~, then compare the pulse counting position indicator and reed switch position indicator channels at least once~~ per 4 hours.

A01

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.1.7

~~3.1.3.2~~ ~~The shutdown and regulating CEA~~ reed switch position indicator channel~~s~~ and ~~CEA~~ pulse counting position indicator channel~~s~~ shall be OPERABLE ~~and capable of determining the absolute CEA positions within ± 2.50 inches.~~ for each CEA

A05

Applicability

APPLICABILITY: MODES 1 and 2.

Separate Condition entry is allowed for each CEA group.

L01

ACTION:

One or more CEA groups with

Condition A

~~a.~~ ~~With a maximum of~~ one reed switch position indicator channel ~~per group~~ or one ~~(except as permitted by ACTION item c. below)~~ pulse counting position indicator channel ~~per group~~ inoperable ~~and the CEA(s) with the~~ inoperable position indicator channel partially inserted, within 6 hours either:

inoperable

A02

Condition A Note

Only applicable when CEA

is associated with

ITS Required Action A.1, A.2, and Note

L04

1. ~~Restore the inoperable position indicator channel to OPERABLE status, or~~

MODE 3

Required Action D.1

2. Be in ~~HOT STANDBY, or~~

RTP

Required Action A.3.1

3. Reduce THERMAL POWER to ≤ 70% ~~of the maximum allowable THERMAL POWER level for the existing Reactor Coolant pump combination; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Operation at or below this reduced THERMAL POWER level may continue provided that~~ within the next 4 hours ~~either:~~

A03

A04

within 10 hours

Required Action A.3.2.1

Required Action A.3.2.1 Note

Required Action B.1

Verify associated

~~a)~~ ~~The~~ CEA ~~group(s)~~ with ~~the~~ inoperable position indicator ~~is~~ fully withdrawn ~~while~~ maintaining the withdrawal sequence required by Specification 3.1.3.6 ~~and when this~~ CEA ~~group reaches its fully withdrawn position, the~~ "Full Out" limit ~~of the CEA with the inoperable~~ position indicator is actuated ~~and verifies this CEA to be fully withdrawn. Subsequent to fully withdrawing this CEA group(s), the THERMAL POWER level may be returned to a level consistent with all other applicable specifications; or~~

A01

A04

within 6 hours [A.2], within 10 hours [A.3.2.2]

Required Action A.3.2.2

Required Action A.3.2.2 Note

Required Action B.2

Verify associated

~~b)~~ ~~The~~ CEA ~~group(s)~~ with the inoperable position indicator ~~is~~ fully inserted, and subsequently maintained fully ~~inserted,~~ while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 ~~and when this~~ CEA ~~group reaches its fully inserted position, the~~ "Full In" limit ~~of the CEA with the inoperable~~ position indicator is actuated ~~and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6.~~

A01

A04

A01

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS - OPERATING

ACTION: (Continued)

Condition B

b. ~~With a maximum of~~ One or more CEA groups with one reed switch position indicator channel ~~per group~~ or one pulse counting position indicator channel ~~per group~~ inoperable ~~and the full-length CEA(s) with the~~ inoperable position indicator channel ~~at either its~~ fully inserted ~~position~~ or ~~fully~~ withdrawn ~~position, operation may continue provided:~~

Condition B Note: Only applicable when inoperable position indicator channel is associated with CEA

Required Action B.1 and B.2: Verify associated

1. ~~The position of an affected full-length~~ CEA ~~is verified~~ immediately and ~~at least~~ once per 12 hours thereafter ~~by its~~ "Full In" or "Full Out" limit ~~(as applicable), and~~ position indicator is actuated

2. ~~The fully inserted full-length CEA group(s) containing the inoperable position indicator channel is subsequently maintained fully inserted, and~~

A06

3. ~~Subsequent operation is within the limits of Specification 3.1.3.6.~~

A04

Condition C

c. One or more groups with two ~~With two or more pulse counting position indicators channels per group inoperable, operation in MODES 1 and 2 may continue for up to~~ 72 hours ~~provided~~ no more than one reed switch position indicator per group is inoperable.

Require Action C.1: Verify per group is inoperable

Require Action C.2: Restore pulse counting position indicator channels to OPERABLE status within

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

4.1.3.2 ~~Each position indicator channel shall be determined to be OPERABLE by~~ verify~~ing the~~ pulse counting position indicator channels and ~~the~~ reed switch position indicator channels agree within 5.0 inches in accordance with the Surveillance Frequency Control Program ~~except during time intervals~~ when ~~the~~ Deviation Circuit is inoperable~~, then compare the pulse counting position indicator and reed switch position indicator channels at least once per~~ 4 hours.

DISCUSSION OF CHANGES
ITS 3.1.7, CONTROL ELEMENT ASSEMBLY (CEA) POSITION INDICATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.1.3.3 ACTION b. covers the inoperability for a maximum of one reed switch position indicator channel per group (except as permitted by ACTION d.). Unit 1 CTS 3.1.3.3 ACTION d. covers the inoperability for one or more pulse counting position indicator channels inoperable and the maximum allowed inoperable reed switch position indicator channels per group to allow continued operation in MODES 1 and 2.

Unit 2 CTS 3.1.3.2 ACTION a. covers the inoperability for a maximum of one reed switch position indicator channel per group except as permitted by ACTION d.). Unit 2 CTS 3.1.3.2 ACTION c. covers the inoperability for one or more pulse counting position indicator channels inoperable and the maximum allowed inoperable reed switch position indicator channels per group to allow continued operation in MODES 1 and 2.

The purpose of the CTS “(except as permitted by ACTION d.)” statement is to alert the user to a limitation on inoperable reed switch position indicator channels per group, when one or more pulse counting indicator channels is inoperable. This changes the CTS by not including the cross-reference in the ITS. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A03 Unit 1 CTS 3.1.3.3 ACTION b.3, and Unit 2 CTS 3.1.3.2 ACTION a.3, each state that THERMAL POWER be reduced to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination, and if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used.

The purpose of the CTS is to provide the THERMAL POWER limit should it be necessary to reduce THERMAL POWER. The ITS retains this requirement but does not contain a method to be used to reduce THERMAL POWER. PSL only operates in the four reactor coolant pump combination; therefore, subsequent operation is allowable provided the THERMAL POWER is reduced to ≤ 70% RTP – the equivalent THERMAL POWER for the CTS four reactor coolant pump combination. Additionally, the method to be used to reduce THERMAL POWER is determined by Licensed Operators based upon plant conditions and without

challenging plant systems. This changes the CTS by changing the statement that THERMAL POWER be reduced to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination to THERMAL POWER be reduced to ≤ 70% RTP, and by removing the method to be used to reduce THERMAL POWER. This changes the ITS by stating the specific THERMAL POWER limit should it be necessary to reduce THERMAL POWER.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A04 Unit 1 CTS 3.1.3.3 ACTIONS b.3, b.3.a, b.3.b, c.3, and Unit 2 CTS 3.1.3.2 ACTIONS a.3, a.3.a, a.3.b, b.3, each contain statements in reference to CTS 3.1.3.6 requirements including; (b.3, a.3) operation at or below this reduced THERMAL POWER level may continue provided requirements in Specification 3.1.3.6 are maintained; (b.3.a, a.3.a) subsequent to fully withdrawing this CEA group(s), the THERMAL POWER level may be returned to a level consistent with all other applicable specifications; (b.3.b, a.3.b) subsequent operation shall be within the limits of Specification 3.1.3.6.; and (c.3, b.3) subsequent operation is within the limits of Specification 3.1.3.6. Each CTS statement is in reference to requirements in CTS 3.1.3.6, “Regulating CEA Insertion Limits.”

The purpose of the CTS is to provide reference to CTS 3.1.3.6 to determine if the CEA Insertion Limits are met. The ITS does not contain these references. This changes the CTS by not including the reference to the CEA insertion limits in the ITS. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

A05 Unit 1 CTS 3.1.3.3 LCO states that shutdown and regulating CEA reed switch position indicator channels and CEA pulse counting position indicator channels shall be OPERABLE and capable of determining the absolute CEA positions within $\pm$ 2.25 inches. Similarly, Unit 2 CTS 3.1.3.2 LCO states that shutdown and regulating CEA reed switch position indicator channels and CEA pulse counting position indicator channels shall be OPERABLE and capable of determining the absolute CEA positions within $\pm$ 2.50 inches.

The purpose of the CTS is to provide acceptance criteria for the accuracy of the CEA reed switch position indicator channels and CEA pulse counting position indicator channels. The acceptance criteria are provided in the ITS 3.1.7 Bases. ITS 3.1.7 Bases LCO description states that for a CEA position indicator channel to be considered OPERABLE, the position indicator must be capable of determining the absolute CEA position within $\pm$ 2.25 inches for Unit 1, and $\pm$ 2.50 inches for Unit 2. ITS Bases Surveillance Requirements, SR 3.1.7.1, description states that CEA pulse counting indicator position channels and reed switch position indicator channels for the CEAs in each group agree within 4.5 inches for Unit 1, and within 5.0 inches for Unit 2. These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A06 Unit 1 CTS 3.1.3.3 Action c.2 states that the fully inserted CEA group(s) containing the inoperable position indicator channel is subsequently maintained fully inserted. Similarly, Unit 2 CTS 3.1.3.3 Action b.2 states that the fully inserted full-length CEA group(s) containing the inoperable position indicator channel is subsequently maintained fully inserted.

The purpose of the CTS is to provide Action to maintain a fully inserted CEA group(s) containing the inoperable position indicator channel in the fully inserted position. For ITS Condition A, with one or more CEA groups with one reed switch or one pulse counting position indicator channel inoperable, Required Actions A.1 and A.2 require fully withdrawing or fully inserting the CEA with the inoperable position indicator. These actions place the CEA in a condition where the position can be verified using the CEA Limits Indication System. The associated CEA "full out" or "full in" limit position indicator actuation provides assurance the CEA is at a known position. Once the CEA is fully withdrawn or fully inserted, Condition A no longer applies, and Condition B is entered.

For ITS Condition B, with one or more CEA groups with one reed switch or one pulse counting position indicator channel inoperable and the associated CEA is fully withdrawn or fully inserted, a verification is required to ensure that either the CEA "full out" or "full in" limit indicator is actuated immediately and every 12 hours thereafter. The associated CEA "full out" or "full in" limit position indicator actuation provides assurance the CEA is at a known position. Use of the CEA limit indicator is an acceptable method to determine CEA position since the CEA Limits Indication System is separate and independent from the reed switch and the pulse counting position indication systems. This change is designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.1.3.3 ACTION b. covers the inoperability for a maximum of one reed switch position indicator channel per group or a maximum of one pulse counting position indicator channel per group with the CEA partially inserted, and CTS 3.1.3.3 ACTION c.

covers the inoperability for a maximum of one rod position indicator channel per group with the CEA fully withdrawn or inserted.
Unit 2 CTS 3.1.3.2 ACTION a. covers the inoperability for a maximum of one reed switch position indicator channel per group or a maximum of one pulse counting position indicator channel per group with the CEA partially inserted, and CTS 3.1.3.2 ACTION b. covers the inoperability for a maximum of one rod position indicator channel per group with the CEA fully withdrawn or inserted.

ITS 3.1.7 ACTIONS are modified by Note 1 that states "Separate Condition entry is allowed for each CEA group." ITS 3.1.7 ACTION A covers inoperability for one or more CEA groups with one reed switch position indicator channel inoperable, or one or more CEA groups with one pulse counting position indicator channel inoperable with the CEA partially inserted. ITS 3.1.7 ACTION B covers inoperability for one or more CEA groups with one reed switch position indicator channel inoperable, or one or more CEA groups with one pulse counting position indicator channel inoperable with the CEA fully withdrawn or inserted.

The purpose of Unit 1 CTS 3.1.3.3 ACTION b. (Unit 2 CTS 3.1.3.2 ACTION a.) is to provide compensatory actions for a maximum of one rod position indicator per group. The purpose of Unit 1 CTS 3.1.3.3 ACTION c. (Unit 2 CTS 3.1.3.2 ACTION b.) is to provide compensatory actions for more than one rod position indicator per group with the CEA fully withdrawn or inserted. This change is acceptable because the Required Actions provide appropriate compensatory actions for one or more inoperable position indicator channels in each CEA group. The actions that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period.

This change will allow separate Condition entry for each inoperable CEA group while the CTS does not. The ITS will allow each inoperable CEA group with an inoperable reed rod position indicator or inoperable pulse counting position indicator to be tracked separately. This change is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for inoperable CEA position indication.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 **Unit 1 only:** *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.3 ACTION d. states “With one or more pulse counting position indicator channels inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided all of the reed switch position indicator channels are OPERABLE.” ITS 3.1.7 Required Action C.1 states “Verify no more than one reed switch position indicator channel per group is inoperable.” This changes the CTS by relaxing the allowable number of inoperable pulse counting position indicator channels and concurrent inoperable reed switch position indicator channels per group from “one or more pulse counting position indicator channels inoperable”

to “one or more CEA groups with two or more pulse counting position indicator channels inoperable”, and from “provided all of the reed switch position indicator channels are OPERABLE” to “no more than one reed switch position indicator channel per group is inoperable.”

The purpose of CTS 3.1.3.3 ACTION d. is to continue to determine CEA position using reed switch position indicator channels, with two or more inoperable pulse counting position indicator channels. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the specified redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 **Unit 1 only:** *(Category 3 – Relaxation of Completion Time)* CTS 3.1.3.3 ACTION d. states “With one or more pulse counting position indicator channels inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided all of the reed switch position indicator channels are OPERABLE.” ITS 3.1.7 Required Action C.2 states that the pulse counting position indicator channels be restored to OPERABLE status within 72 hours. This changes the CTS by relaxing the Completion Time from 24 hours to 72 hours.

This change is acceptable because the Completion Time is consistent with safe operation under the specific Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, and the low probability of a DBA occurring during the allowed Completion Time. This ITS Completion Time of 72 hours is adequate time to restore the inoperable pulse counting position indicator channels to OPERABLE status, considering the reed switch position indicator channels will continue to provide indication of CEA position.

This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L04 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.1.3.3 ACTION b. covers the inoperability for a maximum of one reed switch position indicator channel per group or a maximum of one pulse counting position indicator channel per group with the CEA partially inserted, and CTS 3.1.3.3 ACTION c. covers the inoperability for a maximum of one rod position indicator channel per group with the CEA fully withdrawn or inserted. Unit 2 CTS 3.1.3.2 ACTION a. covers the inoperability for a maximum of one reed switch position indicator channel per group or a maximum of one pulse counting position indicator channel per group with the CEA partially inserted, and CTS 3.1.3.2 ACTION b. covers the inoperability for a maximum of one rod position indicator channel per group with the CEA fully withdrawn or inserted.

ITS 3.1.7 adds Required Action A.1, which states “Fully withdraw CEA with inoperable position indicator channel” with a Completion Time or 6 hours, and adds Required Action A.2, which states “Fully insert CEA with inoperable position indicator channel” with a Completion Time or 6 hours. Unit 1 CTS 3.1.3.3 ACTION b. and Unit 2 CTS 3.1.3.2 ACTION a. are changed by adding an option to fully insert or fully withdraw the affected individual CEA, rather than the CEA group, provided the CEA alignment is maintained in accordance with LCO 3.1.4, “Control Element Assembly Alignment” as annotated in the ITS 3.1.7 Required Action A.1 and Required Action A.2 Note.

The purpose of the Unit 1 CTS 3.1.3.3 ACTION b. (Unit 2 CTS 3.1.3.2 ACTION a. and b.) is to provide compensatory actions for a maximum of one rod position indicator per group inoperable with the CEA partially inserted. The purpose of Unit 1 CTS 3.1.3.3 ACTION c. (Unit 2 CTS 3.1.3.2 ACTION b.) is to provide compensatory actions for a maximum of one rod position indicator channel per group inoperable with the CEA fully withdrawn or inserted. The CTS Required Actions are not affected. Rather, the individual CEA with the inoperable position indicator channel may be moved to the fully withdrawn or fully inserted position rather than moving its CEA group, provided the that contains the individual CEA with the inoperable position indicator channel is maintained in accordance with LCO 3.1.4, “Control Element Assembly Alignment.” The Completion Time is the same as the Completion Time allowed by CTS for alignment of the CEA group with an inoperable position indicator channel.

This change is acceptable because the ITS Required Actions A.1 and A.2 provide appropriate compensatory actions for an inoperable CEA position indication. ITS Required Action A.1 or A.2 may be taken when the CEA group is within the LCO 3.1.4 group alignment limit of the fully inserted or fully withdrawn position. These ACTIONS minimize CEA movements and reduce the likelihood that a reduction in THERMAL POWER. Upon Completion of Required Action A.1 or A.2, Condition A is exited, and Condition B is entered and ACTIONS consistent with the CTS ACTIONS are taken. When the CEA group with the inoperable position indicator channel is not within the LCO 3.1.4 group alignment limit of the fully inserted or fully withdrawn position, then the ITS 3.1.7 Required Action A.3.1 to reduce THERMAL POWER within 6 hours, and Required Action A.3.2.1 to fully withdraw the CEA group with the inoperable position indicator channel within 10 hours, or Required Action A.3.2.2 to fully insert the CEA group with the inoperable position indicator channel within 10 hours, are taken. ITS Required Actions A.1 and A.2 provide appropriate compensatory actions for one inoperable position indicator channel within a CEA group. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Control Element Assembly (CEA) Position Indication

LCO 3.1.7 Reed switch position indicator channel and pulse counting position indicator channel for each CEA shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each CEA group.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------- NOTE ---------- Only applicable when inoperable position indicator channel is associated with partially inserted CEA. ---------- One or more CEA groups with one reed switch position indicator channel inoperable. OR One or more CEA groups with one pulse counting position indicator channel inoperable.	----------NOTE---------- Required Actions A.1 and A.2 are only applicable while maintaining CEA alignment in accordance with LCO 3.1.4, “Control Element Assembly (CEA) Alignment.” ---------- A.1 Fully withdraw CEA with inoperable position indicator channel. OR A.2 Fully insert CEA with the inoperable position indicator channel. OR A.3.1 Reduce THERMAL POWER to ≤ 70% RTP. AND	 6 hours 6 hours 6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	----------NOTE---------- Required Actions A.3.2.1 and A.3.2.2 are only applicable while maintaining CEA group insertion and withdrawal sequence limits specified in the COLR. ----------	
	A.3.2.1 Fully withdraw CEA group with inoperable position indicator channel.	10 hours
	OR	
	A.3.2.2 Fully insert CEA group with the inoperable position indicator channel.	10 hours
B. ---------- NOTE ---------- Only applicable when inoperable position indicator channel is associated with fully withdrawn or inserted CEA. ---------- One or more CEA groups with one reed switch position indicator channel inoperable. OR One or more CEA groups with one pulse counting position indicator channel inoperable.	B.1 Verify associated CEA “full out” limit position indicator is actuated. OR B.2 Verify associated CEA “full in” limit position indicator is actuated.	Immediately AND Once per 12 hours thereafter Immediately AND Once per 12 hours thereafter

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more CEA groups with two or more pulse counting position indicator channels inoperable.	C.1 Verify no more than one reed switch position indicator channel per group is inoperable. AND C.2 Restore pulse counting position indicator channels to OPERABLE status.	Immediately 72 hours
D. One or more CEA groups with two or more reed switch position indicator channels inoperable. OR Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify CEA pulse counting position indicator channels and reed switch position indicator channels agree within 4.5 inches.	4 hours when deviation circuit is inoperable AND In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Control Element Assembly (CEA) Position Indication

LCO 3.1.7 Reed switch position indicator channel and pulse counting position indicator channel for each CEA shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each CEA group.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------- NOTE ---------- Only applicable when inoperable position indicator channel is associated with partially inserted CEA. ---------- One or more CEA groups with one reed switch position indicator channel inoperable. <u>OR</u> One or more CEA groups with one pulse counting position indicator channel inoperable.	----------NOTE---------- Required Actions A.1 and A.2 are only applicable while maintaining CEA alignment in accordance with LCO 3.1.4, “Control Element Assembly (CEA) Alignment.” ---------- A.1 Fully withdraw CEA with inoperable position indicator channel. <u>OR</u> A.2 Fully insert CEA with the inoperable position indicator channel. <u>OR</u> A.3.1 Reduce THERMAL POWER to ≤ 70% RTP. <u>AND</u>	 6 hours 6 hours 6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	----------NOTE---------- Required Actions A.3.2.1 and A.3.2.2 are only applicable while maintaining CEA group insertion and withdrawal sequence limits specified in the COLR. ----------	
	A.3.2.1 Fully withdraw CEA group with inoperable position indicator channel.	10 hours
	OR	
	A.3.2.2 Fully insert CEA group with the inoperable position indicator channel.	10 hours
B. ---------- NOTE ---------- Only applicable when inoperable position indicator channel is associated with fully withdrawn or inserted CEA. ---------- One or more CEA groups with one reed switch position indicator channel inoperable. OR One or more CEA groups with one pulse counting position indicator channel inoperable.	B.1 Verify associated CEA “full out” limit position indicator is actuated. OR B.2 Verify associated CEA “full in” limit position indicator is actuated.	Immediately AND Once per 12 hours thereafter Immediately AND Once per 12 hours thereafter

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more CEA groups with two or more pulse counting position indicator channels inoperable.	C.1 Verify no more than one reed switch position indicator channel per group is inoperable. AND C.2 Restore pulse counting position indicator channels to OPERABLE status.	Immediately 72 hours
D. One or more CEA groups with two or more reed switch position indicator channels inoperable. OR Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify CEA pulse counting position indicator channels and reed switch position indicator channels agree within 5.0 inches.	4 hours when deviation circuit is inoperable AND In accordance with the Surveillance Frequency Control Program

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.7, CONTROL ELEMENT ASSEMBLY (CEA) POSITION INDICATION

1. NUREG 1432, “Standard Technical Specifications, Combustion Engineering Plants”, Revision 5.0, ISTS 3.1.4 includes one Surveillance Requirement (SR) related to CEA position indication, ISTS SR 3.1.4.5. However, the LCO statement of ISTS 3.1.4 does not specifically delineate position indication as being required for a CEA. It only requires OPERABILITY of CEAs to be within the alignment limits. Furthermore, the LCO Section of the ISTS 3.1.4 Bases does not describe indication as a necessary feature for the CEA to be OPERABLE. However, the ISTS Bases for SR 3.1.4.5 provides a requirement that each reed switch position indication channel ensures the channel is OPERABLE and capable of indicating CEA position over the entire length of the CEA's travel. In addition, ISTS 3.1.4 does not provide any specific actions to take if one or more of the required CEA position indicators is inoperable. Consistent with the approach taken in NUREG 1431, “Standard Technical Specifications, Westinghouse Plants”, Revision 5.0, a new LCO has been provided for the CEA Position Indicators.

   PSL Unit 1 CTS 3.1.3.3 and PSL Unit 2 CTS 3.1.3.2 are adapted using NUREG 1431, ISTS 3.1.7, to develop the new LCO 3.1.7. PSL LCO 3.1.7 maintains the current licensing basis while adapting the new LCO to NUREG 1431 requirements. Specifically, PSL requires two CEA position indicator channels for each CEA to be OPERABLE; the reed switch position indicator channels and the pulse counting indicator channel. The ACTIONS are adapted from the ACTIONS in NUREG-1431. The Surveillance Requirement ensures the reed switch position indicator channels and the pulse counting indicator channels for a group agree within an allowable tolerance.

   Refer to the PSL Unit 1 CTS 3.1.3.3 and PSL Unit 2 CTS 3.1.3.2 markups and discussion of changes for the details of changes from CTS to ITS 3.1.7.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Control Element Assembly (CEA) Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. The purpose of this LCO is to ensure OPERABILITY of the shutdown and regulating CEA position indicators so CEA position can be determined thereby ensuring compliance with the CEA alignment and insertion limits.

The OPERABILITY (i.e., trippability) of the shutdown and regulating CEAs is an initial assumption in the safety analyses that assume CEA insertion upon reactor trip. Maximum CEA misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10 and GDC 26 (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Limits on CEA alignment and OPERABILITY have been established, and CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CEAs are moved by their control element drive mechanisms (CEDMs). Each CEDM moves its CEA one step (approximately 0.75 inches) at a time. Step in and step out signals are totalized within the CEA Control System to provide one of the means of determining CEA position.

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems: Pulse Counting Position Indication System and the Reed Switch Position Indication System. Additionally, the CEA Limits Indication System independently provides indication of a CEA that is fully withdrawn or fully inserted.

BASES

BACKGROUND (continued)

The Pulse Counting Position Indication System counts the pulses sent to the CEA gripper coils from the CEDM Control System that moves the CEAs. There is a step counter for each CEA. Individual CEAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the pulse counter. The Pulse Counting Position Indication System is considered highly precise (± 1 step or ± 0.75 inches).

The Reed Switch Position Indication System provides an independent and diverse means of providing indication of actual CEA position, but at a lower precision than the step counters. This system is based on signals from a series of magnetically actuated reed switch position transmitters spaced at 1.5-inch intervals along the CEDM housing. The reed switch position transmitters provide voltage signals for CEA position monitoring in the control room. The reed switch position transmitters also provide signals for alarm information and CEA motion inhibit on CEA deviation within a group.

The CEA Limits Indication System provides indication of a fully withdrawn or fully inserted CEA from distinct contact closure signals from the reed switch position transmitter assembly. The limit reed switch position transmitter assembly, on each CEA, transmits an upper electrical limit signal when the CEA is fully withdrawn and transmits a lower electrical limit signal when the CEA is fully inserted. The CEA Limits Indication System is separate from the Reed Switch Position Indication System.

A detailed description of the CEA position indication systems is found in the UFSAR, Chapter 7 (Ref. 3)

APPLICABLE SAFETY ANALYSES

CEA misalignment accidents are analyzed in the safety analysis (Ref. 4). The accident analysis defines CEA misoperation as any event, with the exception of sequential group withdraws, which could result from a single malfunction in the reactivity control systems. For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDM Control System, or by operator error. Therefore, the acceptance criteria for CEA position indication is that CEA positions must be known with sufficient accuracy in order to verify the core is operating within the limits of LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits". The CEA positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Control Element Assembly (CEA) Alignment"). CEA positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The position of the CEAs satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). The CEA position indication systems monitor CEA position, which is an initial condition of the accident.

LCO

OPERABILITY of the CEA reed switch and pulse counting position indicator channels ensure that the CEA alignment assumptions in the safety analysis will remain valid. The CEA alignment requirements specified in LCO 3.1.4 ensure that the CEA groups maintain the correct power distribution and CEA alignment.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned CEAs can be detected. Therefore, power peaking and SDM can be controlled within acceptable limits.

For a CEA position indicator channel to be considered OPERABLE, the position indicator must be capable of determining the absolute CEA position within $\pm$ 2.25 inches.

APPLICABILITY

The requirements on CEA position indicator channels are applicable in MODES 1 and 2 because these are the MODES power is generated, and the OPERABILITY and alignment of CEAs and CEA insertion limits have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and regulating banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each CEA group. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for one or more inoperable position indicator channels in each CEA group.

Conditions A and B are modified by a Note. The Note to Condition A indicates that the Condition only applies when the inoperable position indicator channel is associated with a partially inserted CEA and the Note to Condition B indicates that the Condition only applies when the inoperable position indicator channel is associated with a fully withdrawn or fully inserted CEA. Because the actions are different for partially inserted CEAs and CEAs that are fully inserted or fully withdrawn, these

BASES

ACTIONS (continued

Notes identify the applicable Condition when a CEA reed switch or pulse counting position indicator channel in one or more CEA groups is inoperable.

A.1 and A.2

With one or more CEA groups with one reed switch or one pulse counting position indicator channel inoperable, Required Actions A.1 and A.2 require fully withdrawing or fully inserting the CEA with the inoperable position indicator. These actions place the CEA in a condition where the position can be verified using the CEA Limits Indication System. The associated CEA “full out” or “full in” limit position indicator actuation provides assurance the CEA is at a known position. Once the CEA is fully withdrawn or fully inserted, Condition A no longer applies and Condition B is entered. The Completion Time for Required Actions A 1 and A.2 is reasonable, based on operating experience, to fully withdraw or fully insert a CEA during power operation.

Required Actions A.1 and A.2 are modified by a Note to restrict these options to only be applicable if the CEA with the inoperable position indicator channel can be maintained within the group alignment limits required by LCO 3.1.4 when the CEA is fully withdrawn or fully inserted. Otherwise, the CEA group must be fully withdrawn or fully inserted in accordance with Required Actions A.3.1, A.3.2.1, and A.3.2.2.

A.3.1, A.3.2.1, and A.3.2.2

If the CEA with an inoperable position indicator channel cannot be maintained within the group alignment limits of LCO 3.1.4 when withdrawing or inserting the CEA, Required Actions A.3.1, A.3.2.1, and A.3.2.2 allow withdrawing or inserting the CEA group to the fully withdrawn or fully inserted position provided THERMAL POWER is reduced to ≤ 70% RTP.

Required Action A.3.1 requires the THERMAL POWER reduction to place the core in a condition where moving a controlling CEA group will not result in core peaking factors exceeding associated limits (Ref. 3).

Required Action A.3.2.1 and A.3.2.2 require either fully withdrawing or fully inserting the CEA group with the inoperable position indicator channel so the position of the affected CEA can be verified using the CEA Limits Indication System.

The Completion Time of Required Action A.3.1 is reasonable, based on operating experience, for reducing power to ≤ 70% RTP from full power conditions without challenging plant systems.

BASES

ACTIONS (continued

The Completion Time of Required Actions A.3.2.1 and A.3.2.2 takes into account the time to reduce THERMAL POWER and the additional time needed to fully withdraw or fully insert the associated CEA group.

Required Actions A.3.2.1 and A.3.2.2 are modified by a Note to only allow fully withdrawing or fully inserting a CEA group with the inoperable position indicator channel if the CEA group can be maintained within the CEA group withdrawal sequence and insertion limits specified in the COLR, as required by LCO 3.1.6.

B.1 and B2

With one or more CEA groups with one reed switch or one pulse counting position indicator channel inoperable and the associated CEA is fully withdrawn or fully inserted, a verification is required to ensure that either the CEA “full out” or “full in” limit indicator is actuated immediately and every 12 hours thereafter. The associated CEA “full out” or “full in” limit position indicator actuation provides assurance the CEA is at a known position. Use of the CEA limit indicator is an acceptable method to determine CEA position since the CEA Limits Indication System is separate and independent from the reed switch and the pulse counting position indication systems. Performing the verification immediately, and every 12 hours thereafter, is considered acceptable in view of other information available to the operator in the control room to determine CEA position and that the actual CEA position is not expected to change.

C.1 and C.2

With one or more CEA groups with two or more pulse counting position indicator channels inoperable, continued operation in MODES 1 and 2 may continue provided verification that no more than one reed switch position indicator channel per group is inoperable. Required Action C.1 ensures no more than one CEA per group has lost position indication. The Completion Time of immediately ensures that prompt action is taken to verify the status of the remaining CEA position indicator channels within the group.

Required Action C.2 requires the pulse counting position indicator channels be restored to OPERABLE status within 72 hours. The Completion Time of 72 hours is adequate time to restore the inoperable pulse counting position indicator channels to OPERABLE status, considering the reed switch position indicators will continue to provide indication of CEA position.

BASES

ACTIONS (continued)

D.1

If two or more reed switch position indicator channels are inoperable in one or more CEA groups, or the Required Actions cannot be completed within the associated Completion Times, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

This SR verifies that CEA pulse counting indicator position channels and reed switch position indicator channels for the CEAs in each group agree within 4.5 inches. This Surveillance is performed every 4 hours when the alignment deviation circuit is inoperable. Performing the verification every 4 hours when the alignment deviation circuit is inoperable, is considered acceptable in view of other information continuously available to the operator in the control room. The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
2. 10 CFR 50.46
3. UFSAR, Chapter 7.5.1.3.
4. UFSAR, Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Control Element Assembly (CEA) Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. The purpose of this LCO is to ensure OPERABILITY of the shutdown and regulating CEA position indicators so CEA position can be determined thereby ensuring compliance with the CEA alignment and insertion limits.

The OPERABILITY (i.e., trippability) of the shutdown and regulating CEAs is an initial assumption in the safety analyses that assume CEA insertion upon reactor trip. Maximum CEA misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10 and GDC 26 (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Limits on CEA alignment and OPERABILITY have been established, and CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CEAs are moved by their control element drive mechanisms (CEDMs). Each CEDM moves its CEA one step (approximately 0.75 inches) at a time. Step in and step out signals are totalized within the CEA Control System to provide one of the means of determining CEA position.

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems: Pulse Counting Position Indication System and the Reed Switch Position Indication System. Additionally, the CEA Limits Indication System independently provides indication of a CEA that is fully withdrawn or fully inserted.

BASES

BACKGROUND (continued)

The Pulse Counting Position Indication System counts the pulses sent to the CEA gripper coils from the CEDM Control System that moves the CEAs. There is a step counter for each CEA. Individual CEAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the pulse counter. The Pulse Counting Position Indication System is considered highly precise (± 1 step or ± 0.75 inches).

The Reed Switch Position Indication System provides an independent and diverse means of providing indication of actual CEA position, but at a lower precision than the step counters. This system is based on signals from a series of magnetically actuated reed switch position transmitters spaced at 1.5-inch intervals along the CEDM housing. The reed switch position transmitters provide voltage signals for CEA position monitoring in the control room. The reed switch position transmitters also provide signals for alarm information and CEA motion inhibit on CEA deviation within a group.

The CEA Limits Indication System provides indication of a fully withdrawn or fully inserted CEA from distinct contact closure signals from the reed switch position transmitter assembly. The limit reed switch position transmitter assembly, on each CEA, transmits an upper electrical limit signal when the CEA is fully withdrawn and transmits a lower electrical limit signal when the CEA is fully inserted. The CEA Limits Indication System is separate from the Reed Switch Position Indication System.

A detailed description of the CEA position indication systems is found in the UFSAR, Chapter 7 (Ref. 3)

APPLICABLE SAFETY ANALYSES

CEA misalignment accidents are analyzed in the safety analysis (Ref. 4). The accident analysis defines CEA misoperation as any event, with the exception of sequential group withdraws, which could result from a single malfunction in the reactivity control systems. For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDM Control System, or by operator error. Therefore, the acceptance criteria for CEA position indication is that CEA positions must be known with sufficient accuracy in order to verify the core is operating within the limits of LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits". The CEA positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Control Element Assembly (CEA) Alignment"). CEA positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The position of the CEAs satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). The CEA position indication systems monitor CEA position, which is an initial condition of the accident.

LCO

OPERABILITY of the CEA reed switch and pulse counting position indicator channels ensure that the CEA alignment assumptions in the safety analysis will remain valid. The CEA alignment requirements specified in LCO 3.1.4 ensure that the CEA groups maintain the correct power distribution and CEA alignment.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned CEAs can be detected. Therefore, power peaking and SDM can be controlled within acceptable limits.

For a CEA position indicator channel to be considered OPERABLE, the position indicator must be capable of determining the absolute CEA position within $\pm$ 2.50 inches.

APPLICABILITY

The requirements on CEA position indicator channels are applicable in MODES 1 and 2 because these are the MODES power is generated, and the OPERABILITY and alignment of CEAs and CEA insertion limits have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and regulating banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each CEA group. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for one or more inoperable position indicator channels in each CEA group.

Conditions A and B are modified by a Note. The Note to Condition A indicates that the Condition only applies when the inoperable position indicator channel is associated with a partially inserted CEA and the Note to Condition B indicates that the Condition only applies when the inoperable position indicator channel is associated with a fully withdrawn or fully inserted CEA. Because the actions are different for partially inserted CEAs and CEAs that are fully inserted or fully withdrawn, these

BASES

ACTIONS (continued

Notes identify the applicable Condition when a CEA reed switch or pulse counting position indicator channel in one or more CEA groups is inoperable.

A.1 and A.2

With one or more CEA groups with one reed switch or one pulse counting position indicator channel inoperable, Required Actions A.1 and A.2 require fully withdrawing or fully inserting the CEA with the inoperable position indicator. These actions place the CEA in a condition where the position can be verified using the CEA Limits Indication System. The associated CEA “full out” or “full in” limit position indicator actuation provides assurance the CEA is at a known position. Once the CEA is fully withdrawn or fully inserted, Condition A no longer applies and Condition B is entered. The Completion Time for Required Actions A 1 and A.2 is reasonable, based on operating experience, to fully withdraw or fully insert a CEA during power operation.

Required Actions A.1 and A.2 are modified by a Note to restrict these options to only be applicable if the CEA with the inoperable position indicator channel can be maintained within the group alignment limits required by LCO 3.1.4 when the CEA is fully withdrawn or fully inserted. Otherwise, the CEA group must be fully withdrawn or fully inserted in accordance with Required Actions A.3.1, A.3.2.1, and A.3.2.2.

A.3.1, A.3.2.1, and A.3.2.2

If the CEA with an inoperable position indicator channel cannot be maintained within the group alignment limits of LCO 3.1.4 when withdrawing or inserting the CEA, Required Actions A.3.1, A.3.2.1, and A.3.2.2 allow withdrawing or inserting the CEA group to the fully withdrawn or fully inserted position provided THERMAL POWER is reduced to ≤ 70% RTP.

Required Action A.3.1 requires the THERMAL POWER reduction to place the core in a condition where moving a controlling CEA group will not result in core peaking factors exceeding associated limits (Ref. 3).

Required Action A.3.2.1 and A.3.2.2 require either fully withdrawing or fully inserting the CEA group with the inoperable position indicator channel so the position of the affected CEA can be verified using the CEA Limits Indication System.

The Completion Time of Required Action A.3.1 is reasonable, based on operating experience, for reducing power to ≤ 70% RTP from full power conditions without challenging plant systems.

BASES

ACTIONS (continued

The Completion Time of Required Actions A.3.2.1 and A.3.2.2 takes into account the time to reduce THERMAL POWER and the additional time needed to fully withdraw or fully insert the associated CEA group.

Required Actions A.3.2.1 and A.3.2.2 are modified by a Note to only allow fully withdrawing or fully inserting a CEA group with the inoperable position indicator channel if the CEA group can be maintained within the CEA group withdrawal sequence and insertion limits specified in the COLR, as required by LCO 3.1.6.

B.1 and B2

With one or more CEA groups with one reed switch or one pulse counting position indicator channel inoperable and the associated CEA is fully withdrawn or fully inserted, a verification is required to ensure that either the CEA “full out” or “full in” limit indicator is actuated immediately and every 12 hours thereafter. The associated CEA “full out” or “full in” limit position indicator actuation provides assurance the CEA is at a known position. Use of the CEA limit indicator is an acceptable method to determine CEA position since the CEA Limits Indication System is separate and independent from the reed switch and the pulse counting position indication systems. Performing the verification immediately, and every 12 hours thereafter, is considered acceptable in view of other information available to the operator in the control room to determine CEA position and that the actual CEA position is not expected to change.

C.1 and C.2

With one or more CEA groups with two or more pulse counting position indicator channels inoperable, continued operation in MODES 1 and 2 may continue provided verification that no more than one reed switch position indicator channel per group is inoperable. Required Action C.1 ensures no more than one CEA per group has lost position indication. The Completion Time of immediately ensures that prompt action is taken to verify the status of the remaining CEA position indicator channels within the group.

Required Action C.2 requires the pulse counting position indicator channels be restored to OPERABLE status within 72 hours. The Completion Time of 72 hours is adequate time to restore the inoperable pulse counting position indicator channels to OPERABLE status, considering the reed switch position indicators will continue to provide indication of CEA position.

BASES

ACTIONS (continued)

D.1

If two or more reed switch position indicator channels are inoperable in one or more CEA groups, or the Required Actions cannot be completed within the associated Completion Times, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

This SR verifies that CEA pulse counting indicator position channels and reed switch position indicator channels for the CEAs in each group agree within 5.0 inches. This Surveillance is performed every 4 hours when the alignment deviation circuit is inoperable. Performing the verification every 4 hours when the alignment deviation circuit is inoperable, is considered acceptable in view of other information continuously available to the operator in the control room. The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
2. 10 CFR 50.46
3. UFSAR, Chapter 7.5.1.3.
4. UFSAR, Chapter 15.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.7, BASES, CONTROL ELEMENT ASSEMBLY (CEA) POSITION INDICATION

1. NUREG 1432, “Standard Technical Specifications, Combustion Engineering Plants”, Revision 5.0, ISTS 3.1.4 includes one Surveillance Requirement (SR) related to CEA position indication, ISTS SR 3.1.4.5. However, the LCO statement of ISTS 3.1.4 does not specifically delineate position indication as being required for a CEA. It only requires OPERABILITY of CEAs to be within the alignment limits. Furthermore, the LCO Section of the ISTS 3.1.4 Bases does not describe indication as a necessary feature for the CEA to be OPERABLE. However, the ISTS Bases for SR 3.1.4.5 provides a requirement that each reed switch position indication channel ensures the channel is OPERABLE and capable of indicating CEA position over the entire length of the CEA's travel. In addition, ISTS 3.1.4 does not provide any specific actions to take if one or more of the required CEA position indicators is inoperable. Consistent with the approach taken in NUREG 1431, “Standard Technical Specifications, Westinghouse Plants”, Revision 5.0, a new LCO has been provided for the CEA Position Indicators.

   PSL Unit 1 CTS 3.1.3.3 and PSL Unit 2 CTS 3.1.3.2 are adapted using NUREG 1431, ISTS 3.1.7 and ISTS 3.1.7 Bases, to develop the new LCO 3.1.7. PSL LCO 3.1.7 maintains the current licensing basis while adapting the new LCO to NUREG 1431 requirements. Specifically, PSL requires two CEA position indicator channels for each CEA to be OPERABLE; the reed switch position indicator channels and the pulse counting indicator channel. The ACTIONS are adapted from the ACTIONS in NUREG-1431. The Surveillance Requirement ensures the reed switch position indicator channels and the pulse counting indicator channels for a group agree within an allowable tolerance.

   Refer to the PSL Unit 1 CTS 3.1.3.3 and PSL Unit 2 CTS 3.1.3.2 markups and discussion of changes for the details of changes from CTS to ITS 3.1.7.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.7, CONTROL ELEMENT ASSEMBLY (CEA) POSITION INDICATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 8

3.1.8, Special Test Exceptions (STE) - MODES 1 and 2

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

SPECIAL TEST EXCEPTIONS

(STE) – MODES 1 and 2

~~GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS~~

LIMITING CONDITION FOR OPERATION

LCO 3.1.8

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

A02

~~b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.~~

Applicability **APPLICABILITY:** MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 are suspended, either:

to less than or equal to the test power plateau within 15 minutes

A02

ACTION A

a. Reduce THERMAL POWER ~~sufficiently to satisfy the requirements of Specification 3.2.1~~, or

Add ITS 3.1.8 ACTION B

M01

ACTION B

b. ~~Be in HOT STANDBY within 6 hours.~~

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

4.10.2.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3, or 3.2.4 are suspended and shall be verified to be within the test power plateau.

~~4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3, or 3.2.4 are suspended.~~

A02

A01

~~SPECIAL TEST EXCEPTIONS~~

~~PRESSURE/TEMPERATURE LIMITATION – REACTOR CRITICALITY~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.3 This specification deleted.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.3 This specification deleted.~~

~~SPECIAL TEST EXCEPTIONS~~

A01

~~PHYSICS TESTS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.4 This specification deleted.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.4 This specification deleted.~~

ITS

A01

SPECIAL TEST EXCEPTIONS

(STE) – MODES 1 and 2

A03

~~3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS~~

LIMITING CONDITION FOR OPERATION

LCO 3.1.8

3.10.2 The moderator temperature coefficient, group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 may be suspended during performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

A02

b. ~~The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.~~

Applicability **APPLICABILITY:** MODES 1 and 2.

A02

to less than or equal to the test power plateau within 15 minutes

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 are suspended, either:

ACTION A

a. Reduce THERMAL POWER ~~sufficiently to satisfy the requirements of Specification 3.2.1~~, or

Add ITS 3.1.8 ACTION B

M01

ACTION B

b. ~~Be in HOT STANDBY within 6 hours~~.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

4.10.2.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1,3.1.3.5, 3.1.3.6, 3.2.3, or 3.2.4 are suspended and shall be verified to be within the test power plateau.

~~4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3, or 3.2.4 are suspended.~~

A02

A01

SPECIAL TEST EXCEPTIONS

(STE) – MODES 1 and 2

~~3/4.10.5 CEA INSERTION DURING ITC, MTC, AND POWER COEFFICIENT MEASUREMENTS~~

A03

LIMITING CONDITION FOR OPERATION

LCO 3.1.8

3.10.5 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS ~~to determine the isothermal temperature coefficient, moderator temperature coefficient, and power coefficient~~ provided the ~~limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.5.2 below.~~

LA01

A02

provided THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP.

M02

Applicability **APPLICABILITY**: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

to less than or equal to the test power plateau within 15 minutes

A02

ACTION A

a. Reduce THERMAL POWER ~~sufficiently to satisfy the requirements of Specification 3.2.1~~, or

Add ITS 3.1.8 ACTION B

M01

ACTION B

b. ~~Be in HOT STANDBY within 6 hours~~.

In accordance with the Surveillance Frequency Control Program

LA02

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

4.10.5.1 The THERMAL POWER shall be determined ~~at least once per hour~~ during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

~~4.10.5.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended.~~

A02

DISCUSSION OF CHANGES
ITS 3.1.8, SPECIAL TEST EXCEPTIONS (STE) – MODES 1 and 2

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.10.2 and Unit 2 CTS 3.10.5 require the linear heat rate (LHR) limits of Specification 3.2.1 to be maintained and determined in accordance with the applicable Surveillances (CTS 4.10.2.2 and 4.10.5.2, respectively). These Surveillances require verifying LHR is within limits by monitoring continuously with the Incore Detector Monitoring System during PHYSICS TESTS above 5% RTP when the applicable LCOs are suspended. Additionally, Action a of CTS 3.10.2 and Unit 2 CTS 3.10.5 require reducing THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1. Neither CTS 3.10.2 nor Unit 2 CTS 3.10.5 allow suspension of CTS 3.2.1 requirements. Therefore, it is unnecessary to state that the LHR requirements of Specification 3.2.1 be maintained and determined to be within limits because CTS 3.2.1 (ITS 3.2.1) will continue to be required in MODE 1 (i.e., above 5% RTP) irrespective of whether PHYSICS TESTS are in progress.

In addition, testing maintains THERMAL POWER within the test power plateau and limits THERMAL POWER to < 85% RTP as specified in CTS 3.10.2.a. ITS LCO 3.1.8 retains this requirement. Limiting THERMAL POWER to < 85% RTP ensures that LHRs are maintained within acceptable limits. Therefore, the limits specified in CTS 3.2.1 (ITS 3.2.1) will be maintained in MODE 1 during PHYSICS TESTS.

This change is acceptable because the requirement to restrict THERMAL POWER to the test power plateau and not to exceed 85% RTP ensures the LHR requirement of Specification 3.2.1 is maintained within the limits specified in the COLR. Therefore, explicitly stating this as a requirement is unnecessary. Additionally, ITS 3.1.8 Required Action A.1, which requires reducing THERMAL POWER to less than or equal to the test power plateau within 15 minutes when the test power plateau is exceeded, ensures THERMAL power will be sufficiently reduced to satisfy the requirements of Specification 3.2.1.

This change is designated as administrative because the ITS requirements achieve the same result as the CTS requirements; maintain the LHR requirements of CTS 3.2.1 within the limits specified in the COLR.

A03 Combines Unit 2 CTS 3.10.2 and CTS 3.10.5 into one Specification, ITS 3.1.8. The requirements of CTS 3.10.5 are encompassed in ITS 3.1.8 by allowing the CEA alignment limits and regulating CEA insertion limits to be suspended to

perform PHYSICS TESTS. Changes related to Unit 2 CTS 3.10.2 are discussed in other Discussion of Changes (DOCs) provided herein.

MORE RESTRICTIVE CHANGES

M01 CTS 3.10.2 and Unit 2 CTS 3.10.5 actions require the unit to be placed in Hot Standby within 6 hours when the limits of Specification 3.2.1 are exceeded while requirements are suspended during PHYSICS TESTS in MODE 1. ITS 3.1.8 requires the PHYSICS TESTS to be suspended within 1 hour when the Required Actions and associated Completion Times are not met. This changes the CTS by requiring the PHYSICS TESTS to be suspended within 1 hour instead of placing the unit in MODE 3 (Hot Standby) within 6 hours. Suspension of the PHYSICS TEST results in canceling the STE and application of the normal Technical Specifications. This change is designated as more restrictive because it requires a more restrictive action within a shorter Completion Time than CTS.

M02 **Unit 2 only:** CTS 3.10.5 allows CTS 3.1.3.1 and 3.1.3.6 to be suspended during performance of PHYSICS TESTS provided the limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.5.2 with no explicit restriction on THERMAL POWER. ITS 3.1.8 also allows suspension of LCO 3.1.4 (CTS 3.1.3.1) and LCO 3.1.6 (CTS 3.1.3.6) requirements provided THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP. This changes the CTS by restricting THERMAL POWER to the test power plateau not to exceed 85% RTP while performing these PHYSICS TESTS. Restricting THERMAL POWER to the test power plateau with a maximum restriction of 85% RTP ensures that adequate LHR and DNB parameter margins are maintained while the applicable LCOs are suspended. Therefore, as discussed in DOC A02 herein, the limits specified in CTS 3.2.1 (ITS 3.2.1) will be maintained in MODE 1 during PHYSICS TESTS.

This change is designated as more restrictive because THERMAL POWER is restricted to the test power plateau with a maximum restriction of 85% RTP.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 **Unit 2 only** (*Type 1 – Removing Details of System Design and System Description, Including Design Limits*) Unit 2 CTS 3.10.5 states, in part, that the purpose of the STE is to "…determine the isothermal temperature coefficient, moderator temperature coefficient, and power coefficient..." ITS does not include the details of why the STE is used. This changes the CTS by eliminating the specific reason the STE is used. R1

The removal of these details, which are related to description of the STE, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate

protection of public health and safety. The ITS explicitly requires THERMAL POWER to be restricted to the test power plateau and not to exceed 85% RTP while performing PHYSICS TEST with the specified LCOs not met. Also, this change is acceptable because the removed information will be adequately controlled in the Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design (i.e., description of the purpose of the STE) is being removed from the Technical Specifications.

LA02 (*Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program*) CTS 4.10.5.1 requires verification that THERMAL POWER be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended and shall be verified to be within the test power plateaus. ITS SR 3.1.8.1 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified periodic Frequency to the Surveillance Frequency Control Program. |R1

The removal of periodic Surveillance Requirement Frequencies is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to periodically verify THERMAL POWER is within the test power plateau. The existing Surveillance Frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. The control of changes to Surveillance Frequencies is in accordance with the Surveillance Frequency Control Program requirements specified in ITS Chapter 5.0. This program ensures that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change because Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Special Test Exceptions (STE) - MODES 1 and 2 ~~(Analog)~~ (1)

3.10.2 LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of:

LCO 3.1.3, "Moderator Temperature Coefficient (MTC),"
LCO 3.1.4, "Control Element Assembly (CEA) Alignment,"
LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits,"
LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits,"
~~LCO 3.2.2, "Total Planar Radial Peaking Factor (F^T_{XY}),"~~
LCO 3.2.~~3~~2, "Total Integrated Radial Peaking Factor (F^T_r)," and
LCO 3.2.~~4~~3, "AZIMUTHAL POWER TILT (T_q)," (2)

may be suspended, provided THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP.

Applicability APPLICABILITY: MODES 1 and 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a DOC A02 A. Test power plateau exceeded.	A.1 Reduce THERMAL POWER to less than or equal to test power plateau.	15 minutes
Action b DOC M01 B. Required Action and associated Completion Time not met.	B.1 Suspend PHYSICS TESTS.	1 hour

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.10.2.1	SR 3.1.8.1	Verify THERMAL POWER is equal to or less than the test power plateau.	~~[1 hour~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 3 3

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Special Test Exceptions (STE) - MODES 1 and 2 ~~(Analog)~~ (1)

3.10.2
3.10.5

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of:

LCO 3.1.3, "Moderator Temperature Coefficient (MTC),"
LCO 3.1.4, "Control Element Assembly (CEA) Alignment,"
LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits,"
LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits,"
~~LCO 3.2.2, "Total Planar Radial Peaking Factor (F^T_{XY}),"~~
LCO 3.2.~~3~~2, "Total Integrated Radial Peaking Factor (F^T_r)," and
LCO 3.2.~~4~~3, "AZIMUTHAL POWER TILT (T_q)," (2)

DOC M02 may be suspended, provided THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP.

Applicability APPLICABILITY: MODES 1 and 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a DOC A02 A. Test power plateau exceeded.	A.1 Reduce THERMAL POWER to less than or equal to test power plateau.	15 minutes
Action b DOC M01 B. Required Action and associated Completion Time not met.	B.1 Suspend PHYSICS TESTS.	1 hour

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.10.2.1 4.10.5.1	SR 3.1.8.1	Verify THERMAL POWER is equal to or less than the test power plateau.	~~[1 hour~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.8, SPECIAL TEST EXCEPTIONS (STE) – MODES 1 and 2

1. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

B 3.1.8 Special Test Exceptions (STE) - MODES 1 and 2 ~~(Analog)~~ 1

BASES

BACKGROUND

The primary purpose of these MODES 1 and 2 Special Test Exceptions (STE) is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine specific reactor core characteristics.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed,

b. Validate the analytical models used in design and analysis,

c. Verify assumptions used for predicting plant response,

d. Ensure that installation of equipment in the facility has been accomplished in accordance with design, and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 4).

BASES

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

Examples of PHYSICS TESTS include determination of critical boron concentration, control element assembly (CEA) group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during a PHYSICS TEST with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 define~~s~~d the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate (LHR) remains within its limit, fuel design criteria are preserved. 2

In this test, the following LCOs are suspended:

LCO 3.1.3, "Moderator Temperature Coefficient (MTC),"
LCO 3.1.4, "Control Element Assembly (CEA) Alignment,"
LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits,"
LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits,"
~~LCO 3.2.2, "Total Planar Radial Peaking Factor (F^T_{xy}),"~~ 3
LCO 3.2.~~3~~2, "Total Integrated Radial Peaking Factor (F^T_r)" and
LCO 3.2.~~4~~3, "AZIMUTHAL POWER TILT (T_q)."

The ~~safety analysis (Ref. 6) places~~ limits on allowable THERMAL POWER during PHYSICS TESTS ~~and requires~~ the LHR and the departure from nucleate boiling (DNB) parameter ~~to be~~ maintained within limits. The power plateau of < 85% RTP and the associated trip setpoints are required to ensure ~~[explain]~~. 2

ensure
are
adequate margin is provided to the LHR and DNB parameter limits so that fuel design limits are not violated during PHYSICS TESTS 4

BASES

APPLICABLE SAFETY ANALYSES (continued)

The individual LCOs governing CEA group height, insertion and alignment, ASI, ~~F^T_{XY}~~, F^T_r, and T_q preserve the LHR limits. Additionally, the LCOs governing Reactor Coolant System (RCS) flow, reactor inlet temperature (T_c), and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the loss of coolant accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. ~~7~~6). The criteria for the loss of forced reactor coolant flow accident are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria. 3 2

During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR and DNB parameter limits may be suspended. The results of the accident analysis are not adversely impacted, however, if LHR and DNB parameters are verified to be within their limits while the LCOs are suspended. Therefore, SRs are placed as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these Surveillances allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are ~~F^T_{XY}~~, F^T_r, T_q, and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR. 3

As described in LCO 3.0.7, compliance with Special Test Exceptions LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

BASES

LCO

This LCO permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of PHYSICS TESTS such as those required to:

a. Measure CEA worth,

b. Determine the reactor stability index and damping factor under xenon oscillation conditions,

c. Determine power distributions for nonnormal CEA configurations,

d. Measure rod shadowing factors, and

e. Measure temperature and power coefficients.

Additionally, it permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient (ITC), MTC, and power coefficient.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.2, LCO 3.2.3 and ~~, and LCO 3.2.4~~ may be suspended during the performance of PHYSICS TESTS, provided THERMAL POWER is restricted to test power plateau, which shall not exceed 85% RTP. 3

APPLICABILITY

This LCO is applicable in MODES 1 and 2 because the reactor must be critical at various THERMAL POWER levels to perform the PHYSICS TESTS described in the LCO section. Limiting the test power plateau to < 85% RTP ensures that LHRs are maintained within acceptable limits.

ACTIONS

A.1

If THERMAL POWER exceeds the test power plateau, THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduction. The 15 minute Completion Time ensures that prompt action shall be taken to reduce THERMAL POWER to within acceptable limits.

B.1 ~~and B.2~~ 3

If Required Action A.1 cannot be completed within the required Completion Time, PHYSICS TESTS must be suspended within 1 hour. Allowing 1 hour for suspending PHYSICS TESTS allows the operator sufficient time to change any abnormal CEA configuration back to within the limits of LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6.

BASES

ACTIONS (continued)

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verifying that THERMAL POWER is equal to or less than that allowed by the test power plateau, as specified in the PHYSICS TEST procedure and required by the safety analysis, ensures that adequate LHR and DNB parameter margins are maintained while LCOs are suspended. ~~[The 1 hour Frequency is sufficient, based on the slow rate of power change and increased operational controls in place during PHYSICS TESTS.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5

~~]~~

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision ~~2~~1, ~~August 1978~~. (January 1977) 2
4. ANSI/ANS-19.6.1-1985, December 13, 1985.
5. FSAR, Chapter ~~[~~14~~]~~. (U) 4 2
6. ~~FSAR, Section [15.3.2.1].~~
7. ~~6~~6. 10 CFR 50.46. 2
8. USFAR, Section 15.2.5

B 3.1 REACTIVITY CONTROL SYSTEMS ~~(Analog)~~

B 3.1.8 Special Test Exceptions (STE) - MODES 1 and 2 ~~(Analog)~~ 1

BASES

BACKGROUND

The primary purpose of these MODES 1 and 2 Special Test Exceptions (STE) is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine specific reactor core characteristics.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed,

b. Validate the analytical models used in design and analysis,

c. Verify assumptions used for predicting plant response,

d. Ensure that installation of equipment in the facility has been accomplished in accordance with design, and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 4).

BASES

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

Examples of PHYSICS TESTS include determination of critical boron concentration, control element assembly (CEA) group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during a PHYSICS TEST with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 define~~s~~d the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate (LHR) remains within its limit, fuel design criteria are preserved. 2

In this test, the following LCOs are suspended:

LCO 3.1.3, "Moderator Temperature Coefficient (MTC),"
LCO 3.1.4, "Control Element Assembly (CEA) Alignment,"
LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits,"
LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits,"
~~LCO 3.2.2, "Total Planar Radial Peaking Factor (F^T_{xy}),"~~ 3
LCO 3.2.~~3~~2, "Total Integrated Radial Peaking Factor (F^T_r)" and
LCO 3.2.~~4~~3, "AZIMUTHAL POWER TILT (T_q)."

The ~~safety analysis (Ref. 6) places~~ limits on allowable THERMAL POWER during PHYSICS TESTS ~~and requires~~ the LHR and the departure from nucleate boiling (DNB) parameter ~~to be~~ maintained within limits. The power plateau of < 85% RTP and the associated trip setpoints are required to ensure ~~[explain]~~. 2

ensure
are
adequate margin is provided to the LHR and DNB parameter limits so that fuel design limits are not violated during PHYSICS TESTS 4

BASES

APPLICABLE SAFETY ANALYSES (continued)

The individual LCOs governing CEA group height, insertion and alignment, ASI, ~~F^T_{XY}~~, F^T_r, and T_q preserve the LHR limits. Additionally, the LCOs governing Reactor Coolant System (RCS) flow, reactor inlet temperature (T_c), and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the loss of coolant accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. ~~7~~6). The criteria for the loss of forced reactor coolant flow accident are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria. 3 2

During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR and DNB parameter limits may be suspended. The results of the accident analysis are not adversely impacted, however, if LHR and DNB parameters are verified to be within their limits while the LCOs are suspended. Therefore, SRs are placed as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these Surveillances allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are ~~F^T_{XY}~~, F^T_r, T_q, and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR. 3

As described in LCO 3.0.7, compliance with Special Test Exceptions LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

BASES

LCO

This LCO permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of PHYSICS TESTS such as those required to:

a. Measure CEA worth,

b. Determine the reactor stability index and damping factor under xenon oscillation conditions,

c. Determine power distributions for nonnormal CEA configurations,

d. Measure rod shadowing factors, and

e. Measure temperature and power coefficients.

Additionally, it permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient (ITC), MTC, and power coefficient.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.2, LCO 3.2.3 and ~~, and LCO 3.2.4~~ may be suspended during the performance of PHYSICS TESTS, provided THERMAL POWER is restricted to test power plateau, which shall not exceed 85% RTP. 3

APPLICABILITY

This LCO is applicable in MODES 1 and 2 because the reactor must be critical at various THERMAL POWER levels to perform the PHYSICS TESTS described in the LCO section. Limiting the test power plateau to < 85% RTP ensures that LHRs are maintained within acceptable limits.

ACTIONS

A.1

If THERMAL POWER exceeds the test power plateau, THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduction. The 15 minute Completion Time ensures that prompt action shall be taken to reduce THERMAL POWER to within acceptable limits.

B.1 ~~and B.2~~ 3

If Required Action A.1 cannot be completed within the required Completion Time, PHYSICS TESTS must be suspended within 1 hour. Allowing 1 hour for suspending PHYSICS TESTS allows the operator sufficient time to change any abnormal CEA configuration back to within the limits of LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6.

BASES

ACTIONS (continued)

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verifying that THERMAL POWER is equal to or less than that allowed by the test power plateau, as specified in the PHYSICS TEST procedure and required by the safety analysis, ensures that adequate LHR and DNB parameter margins are maintained while LCOs are suspended. ~~[The 1 hour Frequency is sufficient, based on the slow rate of power change and increased operational controls in place during PHYSICS TESTS.~~ 4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 5

~~]~~

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision ~~2~~1, ~~August 1978~~. (January 1977) 2
4. ANSI/ANS-19.6.1-1985, December 13, 1985.
5. FSAR, Chapter ~~[~~14~~]~~. (U) 4 2
6. ~~FSAR, Section [15.3.2.1].~~
7. ~~6~~6. 10 CFR 50.46. 2
7. USFAR, Section 15.3

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.8 BASES, SPECIAL TEST EXCEPTIONS (STE) – MODES 1 and 2

1. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. Changes are made to be consistent with changes made to the Specification.

4. The ISTS contains bracketed information and/or values that are generic to Westinghouse vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.8, SPECIAL TEST EXCEPTIONS (STE) – MODES 1 and 2

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 9

Relocated/Deleted Current Technical Specifications

CTS 3/4.1.1.3, Boron Dilution

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

REACTIVITY CONTROL SYSTEMS

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant to the reactor pressure vessel < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

R01

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be determined to be ≥ 3000 gpm within one hour prior to the start of and in accordance with the Surveillance Frequency Control Program during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 3000 gpm to the reactor pressure vessel.

REACTIVITY CONTROL SYSTEMS

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant to the reactor pressure vessel < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

R01

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be determined to be ≥ 3000 gpm within 1 hour prior to the start of and in accordance with the Surveillance Frequency Control Program during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 3000 gpm to the reactor pressure vessel.

DISCUSSION OF CHANGES
CTS 3/4.1.1.3, BORON DILUTION

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R01 CTS 3/4.1.1.3 establishes a minimum flow rate to the reactor pressure vessel (RPV) of at least 3000 gpm whenever a reduction in Reactor Coolant System (RCS) boron concentration is being made. This minimum flowrate provides adequate mixing, prevents stratification, and ensures that reactivity changes will be gradual during boron concentration changes in the RCS.

A minimum boron dilution flow rate to the RPV is not assumed as an initial condition of a design basis accident (DBA) or transient analysis nor is it an assumed value to mitigate a DBA or transient (e.g., a boron dilution event). The ITS does not include this Specification. This changes the CTS by relocating this Specification to the Technical Requirements Manual (TRM).

This change is acceptable because the CTS 3/4.1.1.3.Specification does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The minimum boron dilution flow rate to the RPV is not an instrumentation system. Therefore, this Specification does not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The minimum boron dilution flow rate to the RPV is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. This Specification specifies a minimum flowrate value that provides adequate mixing, prevents stratification, and ensures that reactivity changes in the core are gradual during boron concentration changes in the RCS. These limits, however, do not reflect initial condition assumptions in a DBA or transient.

3. The minimum boron dilution flow rate to the RPV is a parameter and not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The minimum boron dilution flow rate to the RPV is a parameter requirement and is not a structure, system, or component which operating experience or

probabilistic risk assessment has shown to be significant to public health and safety.

Since 10 CFR 50.36(c)(2)(ii) criteria have not been satisfied, the Boron Dilution Specification may be relocated to a licensee controlled document outside the Technical Specifications. Control element assembly insertion limits during power operation, shutdown margin requirements during shutdown, and boration concentration requirements during refueling are retained in separate Technical Specifications and ensure adequate excess negative core reactivity is available in the event of an inadvertent boron dilution event. Additionally, RCS circulation requirements to provide mixing and prevent stratification are retained in separate Technical Specifications (i.e., shutdown cooling specifications).

Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specifications did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and have been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.1.3, BORON DILUTION

There are no specific No Significant Hazards Considerations for this Specification.

CTS 3/4.1.2, Boration Systems

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATHS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from the boric acid makeup tank via either a boric acid pump or a gravity feed connection and any charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump* to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

R01

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes** until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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* The flow path from the RWT to the RCS via a single HPSI pump shall only be established if: (a) the RCS pressure boundary does not exist, or (b) RCS pressure boundary integrity exists and no charging pumps are operable. In the latter case, all charging pumps shall be disabled.

** Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

R01

b. At least once per 24 hours, when the Reactor Auxiliary Building air temperature is less than 55°F, by verifying that the Boric Acid Makeup Tank solution temperature is greater than 55°F, when the flow path from the Boric Acid Makeup Tank is required to be OPERABLE.

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REACTIVITY CONTROL SYSTEMS

FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a boric acid makeup pump through a charging pump to the Reactor Coolant System.

b. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a gravity feed valve through a charging pump to the Reactor Coolant System.

c. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

OR

R01

At least two of the following three boron injection flow paths shall be OPERABLE:

d. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both boric acid makeup pumps through a charging pump to the Reactor Coolant System.

e. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both gravity feed valves through a charging pump to the Reactor Coolant System.

f. The flow path from the refueling water storage tank, via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.2 at 200°F; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

R01

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation Signal.

c. At least once per 24 hours when the Reactor Auxiliary Building air temperature is below 55°F by verifying that the solution temperature of the Boric Acid Makeup Tank(s) is above 55°F.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or high pressure safety injection pump* in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump* OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes** until at least one of the required pumps is restored to OPERABLE status.

R01

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least one of the above required pumps shall be demonstrated OPERABLE by verifying the charging pump develops a flow rate of greater than or equal to 40 gpm or the high pressure safety injection pump develops a total head of greater than or equal to 2571 ft. when tested pursuant to the INSERVICE TESTING PROGRAM.

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* The flow path from the RWT to the RCS via a single HPSI pump shall be established only if: (a) the RCS pressure boundary does not exist, or (b) RCS pressure boundary integrity exists and no charging pumps are operable. In the latter case, all charging pumps shall be disabled.

** Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying that each pump develops a flow rate or greater than or equal to 40 gpm when tested pursuant to the INSERVICE TESTING PROGRAM.

R1

REACTIVITY CONTROL SYSTEMS

BORIC ACID PUMPS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid pump shall be OPERABLE if only the flow path through the boric acid pump in Specification 3.1.2.1a above, is OPERABLE.

APPLICABILITY: MODES 5 and 6.

R01

ACTION:

With no boric acid pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one boric acid pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required boric acid pump shall be demonstrated OPERABLE by verifying that the pump develops the specified discharge pressure when tested pursuant to the INSERVICE TESTING PROGRAM.

* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

BORIC ACID PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

R01

With one boric acid pump required for boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid pump(s) shall be demonstrated OPERABLE by verifying that the pump(s) develop the specified discharge pressure when tested pursuant to the INSERVICE TESTING PROGRAM.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank with a minimum borated water volume of 3650 gallons of 3.0 to 3.5 weight percent boric acid (5245 to 6119 ppm boron).

b. The refueling water tank with:

1. A minimum contained volume of 125,000 gallons,

2. A minimum boron concentration of 1900 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving positive reactivity changes* until at least one borated water source is restored to OPERABLE status.

R01

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the boron concentration of the water,

2. Verifying the water level of the tank, and.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the site ambient air temperature is < 40°F.

c. At least once per 24 hours when the Reactor Auxiliary Building air temperature is less than 55°F by verifying that the Boric Acid Makeup Tank solution temperature is greater than 55°F when that Boric Acid Makeup Tank is required to be OPERABLE.

* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following four borated water sources shall be OPERABLE:

a. Boric Acid Makeup Tank 1A in accordance with Figure 3.1-1.

b. Boric Acid Makeup Tank 1B in accordance with Figure 3.1-1.

c. Boric Acid Makeup Tanks 1A and 1B with a minimum combined contained borated water volume in accordance with Figure 3.1-1.

d. The refueling water tank with:

1. A minimum contained volume of 477,360 gallons of water,

2. A minimum boron concentration of 1900 ppm,

3. A maximum solution temperature of 100°F,

4. A minimum solution temperature of 55°F when in MODES 1 and 2, and

5. A minimum solution temperature of 40°F when in MODES 3 and 4.

R01

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.2 at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two borated water sources shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the boron concentration of the water source,

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

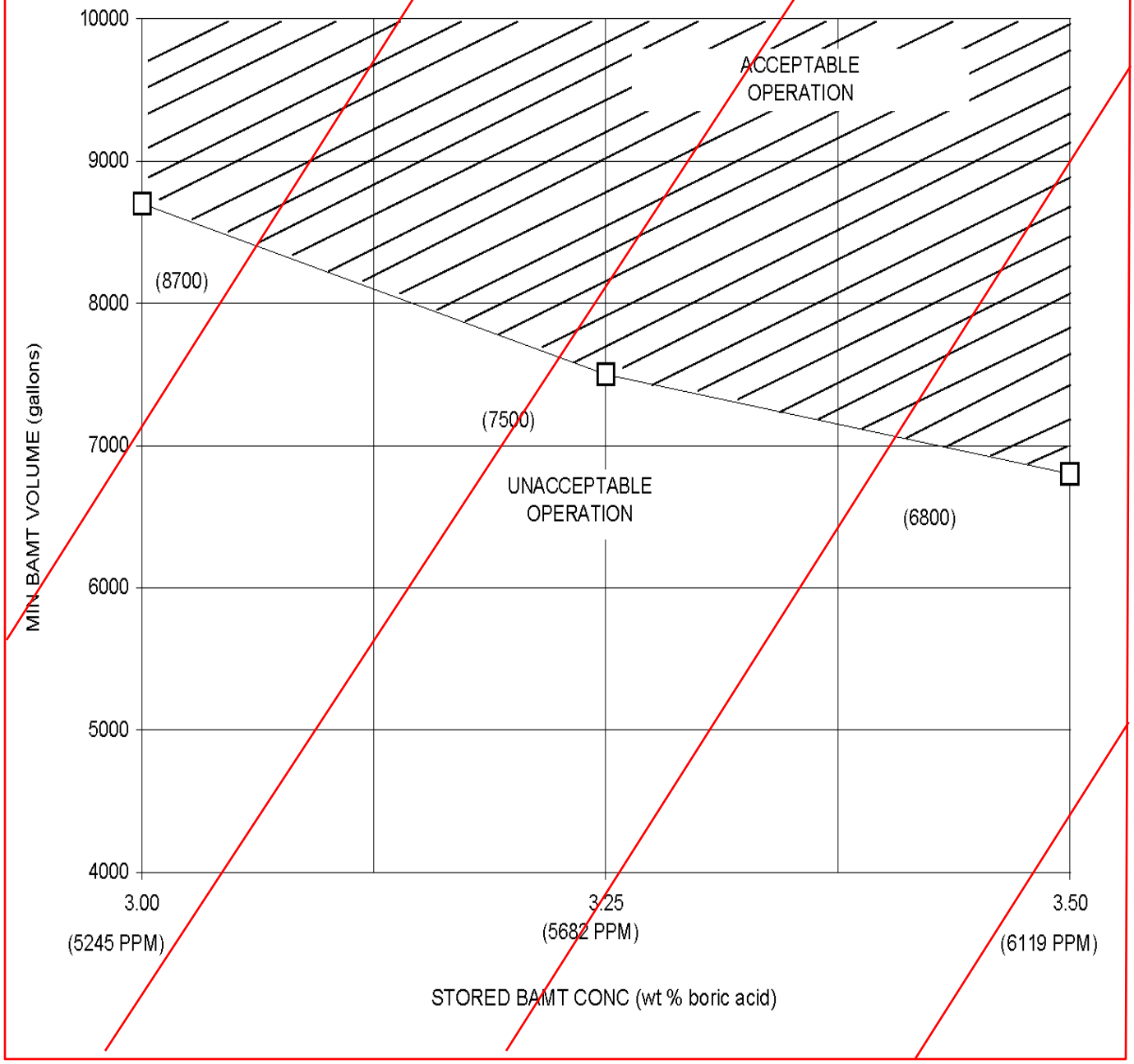
R01

2. Verifying the water level in each water source.

b. In accordance with the Surveillance Frequency Control Program by verifying the RWT temperature.

c. At least once per 24 hours by verifying that the Boric Acid Makeup Tank solution temperature is greater than 55°F when the Reactor Auxiliary Building air temperature is below 55°F.

FIGURE 3.1-1 ST. LUCIE 1 MIN BAMT VOLUME VS STORED BAMT CONCENTRATION (MODES 1, 2, 3 and 4)



REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATHS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid makeup tank via either a boric acid makeup pump or a gravity feed connection and any charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification 3.1.2.7a. is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

R01

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes*.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 24 hours when the Reactor Auxiliary Building air temperature is less than 55°F by verifying that the Boric Acid Makeup Tank solution temperature is greater than 55°F (when the flow path from the Boric Acid Makeup Tank is used).

* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a boric acid makeup pump through a charging pump to the Reactor Coolant System.

b. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a gravity feed valve through a charging pump to the Reactor Coolant System.

c. The flow path from the refueling water tank via a charging pump to the Reactor Coolant System.

OR

At least two of the following three boron injection flow paths shall be OPERABLE: R01

d. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both boric acid makeup pumps through a charging pump to the Reactor Coolant System.

e. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both gravity feed valves through a charging pump to the Reactor Coolant System.

f. The flow path from the refueling water tank, via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200 °F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

R01

a. At least once per 24 hours, when the Reactor Auxiliary Building air temperature is below 55°F, by verifying that the solution temperature of the Boric Acid Makeup Tanks is above 55°F.

b. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. In accordance with the Surveillance Frequency Control Program during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on an SIAS test signal.

d. In accordance with the Surveillance Frequency Control Program by verifying that the flow path required by Specification 3.1.2.2a and 3.1.2.2b delivers at least ~~40~~ gpm to the Reactor Coolant System.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or high pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes*.

R01

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least the above required pumps shall be demonstrated OPERABLE by verifying the charging pump develops a flow rate of greater than or equal to 40 gpm or the high pressure safety injection pump develops a total head of greater than or equal to 2854 ft. when tested pursuant to the INSERVICE TESTING PROGRAM.

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* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

R01

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying that each pump develops a flow rate of greater than or equal to 40 gpm when tested pursuant to the INSERVICE TESTING PROGRAM.

4.1.2.4.2 In accordance with the Surveillance Frequency Control Program verify that each charging pump starts automatically on an SIAS test signal.

REACTIVITY CONTROL SYSTEMS

BORIC ACID MAKEUP PUMPS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid makeup pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid pump in Specification 3.1.2.1a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes*.

R01

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required boric acid makeup pump shall be demonstrated OPERABLE by verifying that the pump develops the specified discharge pressure when tested pursuant to the INSERVICE TESTING PROGRAM.

* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

BORIC ACID MAKEUP PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2 is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2 operable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

R01

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid makeup pump(s) shall be demonstrated OPERABLE by verifying that the pump(s) develop the specified discharge pressure when tested pursuant to the INSERVICE TESTING PROGRAM.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank with a minimum borated water volume of 3550 gallons of 3.1 to 3.5 weight percent boric acid (5420 to 6119 ppm boron).

b. The refueling water tank with:

1. A minimum contained borated water volume of 125,000 gallons,

2. A minimum boron concentration of 1900 ppm, and

3. A solution temperature between 40°F and 120°F.

R01

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes*.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume of the tank, and

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the outside air temperature is outside the range of 40°F and 120°F.

.

c. At least once per 24 hours when the Reactor Auxiliary Building air temperature is less than 55°F, by verifying that the boric acid makeup tank solution temperature is greater than 55°F when that boric acid makeup tank is required to be OPERABLE.

* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following four borated water sources shall be OPERABLE:

a. Boric Acid Makeup Tank 2A in accordance with Figure 3.1-1.

b. Boric Acid Makeup Tank 2B in accordance with Figure 3.1-1.

c. Boric Acid Makeup Tanks 2A and 2B with a minimum combined contained borated water volume in accordance with Figure 3.1-1.

d. The refueling water tank with:

1. A minimum contained borated water volume of 477,360 gallons,

2. A boron concentration of between 1900 and 2200 ppm of boron, and

3. A solution temperature of between 55°F and 100°F.

R01

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank(s) inoperable, restore the tank(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200°F; restore the above required boric acid makeup tank(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two required borated water sources shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

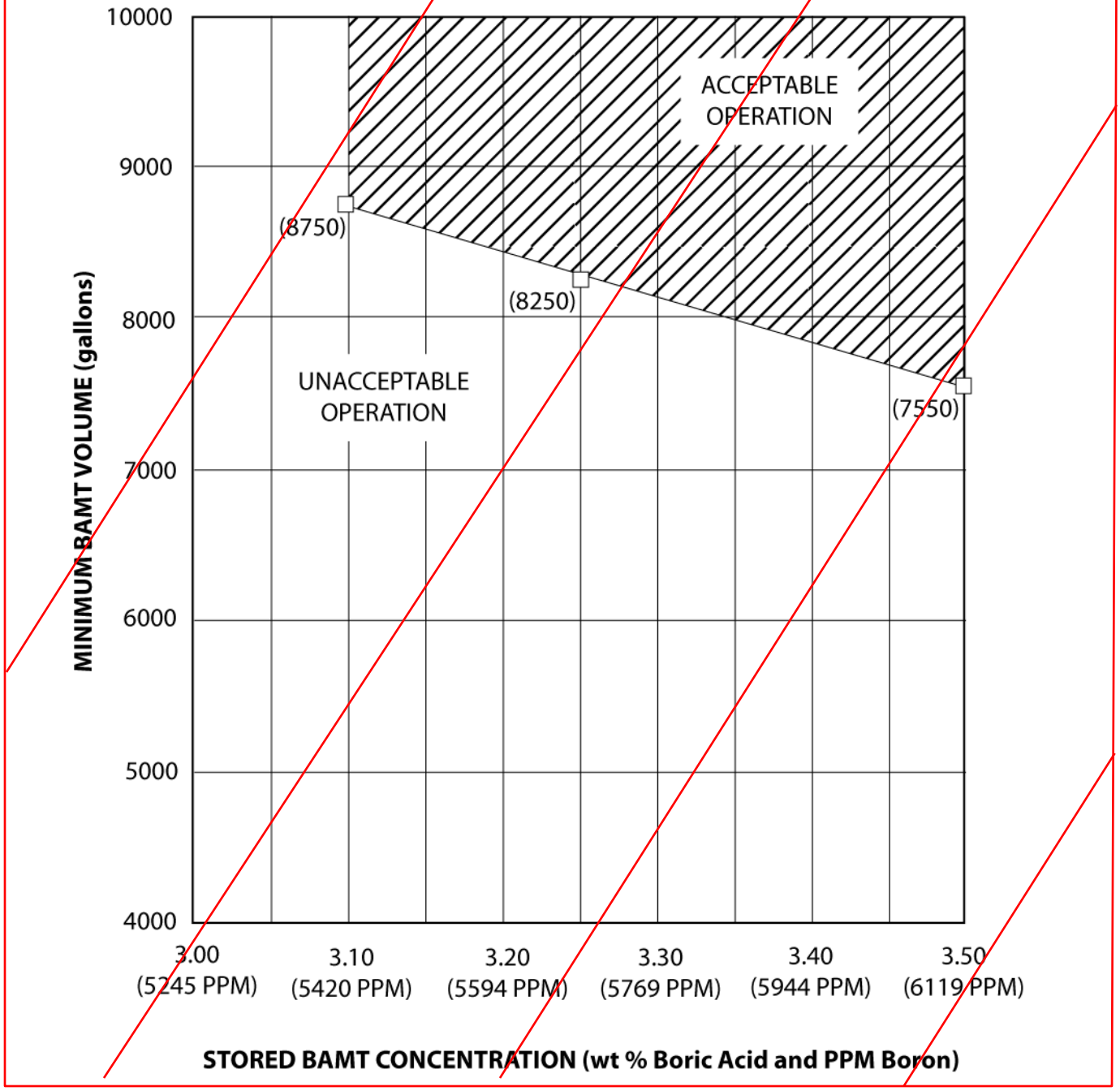
1. Verifying the boron concentration in the water and

2. Verifying the contained borated water volume of the water source.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is outside the range of 55°F and 100°F.

c. At least once per 24 hours when the Reactor Auxiliary Building air temperature is less than 55°F, by verifying that the boric acid makeup tank solution is greater than 55°F.

FIGURE 3.1-1
MINIMUM BAMT VOLUME vs STORED BORIC ACID CONCENTRATION



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DISCUSSION OF CHANGES
CTS 3/4.1.2, BORATION SYSTEMS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R01 CTS 3/4.1.2.1, 3/4.1.2.2, 3/4.1.2.3, 3/4.1.2.4, 3/4.1.2.5, 3/4.1.2.6, 3/4.1.2.7, and 3/4.1.2.8 provide boration system requirements and require a source of borated water, one or more flow paths to inject borated water into the RCS, and either a charging pump or high pressure safety injection pump to provide the necessary charging head.

The purpose of the boration system Technical Specifications is provide the means to control the chemical neutron absorber (boron) concentration in the Reactor Coolant System (RCS) and to help maintain the shutdown margin during normal operations. The RCS boration management system functions to control boron concentration and maintain shutdown margin are not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of a component in the boration systems which causes a boron dilution event, the automatic response, or that required by the operator, is to close the appropriate valves in the reactor makeup system. The plant response to a boron dilution event also includes control rod assembly movement and reactor trip features to ensure shutdown margin is maintained. The boration capabilities of the boration systems are not assumed to mitigate the boron dilution event. The ITS does not include these Specifications. This changes the CTS by relocating these Specifications to the Technical Requirements Manual (TRM).

This change is acceptable because the CTS 3/4.1.2.boration system Specifications do not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The boration systems do not constitute instrumentation systems that are used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The boration systems are not process variables, design features, or operating restrictions that represent an initial condition of a DBA or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

3. The RCS boration management system functions to control boron concentration and maintain shutdown margin do not represent a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The St. Lucie Plant (PSL) at-power PRA has shown that the RCS boration management system functions are not significant risk contributors to core damage frequency and offsite releases. PSL does not have shutdown PRA model. However, operational experience has shown that the boration management system is not a constraint of prime importance in the mitigation of any accident or transient that results in challenging public health and safety. Therefore, the RCS boration management system functions to control boron concentration and maintain shutdown margin do not represent structures, systems, or components which operating experience or probabilistic risk assessment has shown to be significant to public health and safety

Since 10 CFR 50.36(c)(2)(ii) criteria have not been satisfied for the RCS boration management system functions to control boron concentration and maintain shutdown margin, the Boration System Specifications may be relocated to a licensee controlled document outside the Technical Specifications. RCS boration structures, systems, or components credited as the primary success path which function or actuate to mitigate a DBA or transient are retained in separate Technical Specifications (e.g., Emergency Core Cooling System). Additionally, shutdown margin requirements during shutdown, and boration concentration requirements during refueling are retained in separate Technical Specifications to ensure adequate excess negative core reactivity is available in the event of an inadvertent boron dilution event.

Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specifications did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and have been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.2, BORATION SYSTEMS

There are no specific No Significant Hazards Considerations for this Specification.

CTS 3/4.1.3.3, Position Indicator Channels – Shutdown
Unit 2 only

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA position indicator channel shall be OPERABLE for each shutdown or regulating CEA not fully inserted.

APPLICABILITY: MODES 3*, 4*, and 5*.

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

R01

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required CEA position indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

* With the reactor trip breakers in the closed position.

DISCUSSION OF CHANGES
CTS 3/4.1.3.3, POSITION INDICATOR CHANNELS – SHUTDOWN
UNIT 2 ONLY

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R01 Unit 2 Current Technical Specifications (CTS) 3/4.1.3.3 provide control element assembly (CEA) position indication requirements when the reactor is shutdown and requires one CEA position indication channel for each withdrawn CEA to be OPERABLE when the reactor trip breakers are closed and one or more CEAs are withdrawn.

The purpose of this requirement is to provide the control room operator with indication of the position of a CEA when the CEA is not fully inserted and perform any related operations that are required. The CEA position indication channels do not provide any automatic function and no operator action assumed in accident or transient analyses (e.g., uncontrolled CEA withdrawal event) is initiated based on CEA position. Shutdown margin requirements ensure adequate excess negative reactivity is available to maintain the reactor subcritical when a CEA with the highest reactivity worth is withdrawn while the reactor is shutdown. The ITS does not include this Specification. This changes the CTS by relocating this Specification to the Technical Requirements Manual (TRM).

This change is acceptable because the Unit 2 CTS 3/4.1.3.3.Specification does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The CEA position monitoring system when the reactor is shutdown is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The CEA position monitoring system when the reactor is shutdown is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

3. The CEA position monitoring system when the reactor is shutdown is not a structure, system, or component that is part of the primary success path or which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The CEA position monitoring system when the reactor is shutdown and the reactor trip breakers are closed is not addressed in the St. Lucie Plant Unit 2 PRA and does not represent a system which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Since 10 CFR 50.36(c)(2)(ii) criteria have not been satisfied for the CEA position monitoring system when the reactor trip breakers are closed and one or more CEAs are withdrawn, the requirements of the CEA Position Indication – Shutdown Specification may be relocated to a licensee controlled document outside the Unit 2 Technical Specifications. Shutdown Margin requirements during shutdown are retained in separate Technical Specifications and ensure adequate excess negative core reactivity is available in the event of a CEA system malfunction during shutdown (e.g., uncontrolled CEA withdrawal event from a subcritical condition).

Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specifications did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and have been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.3.3, POSITION INDICATOR CHANNELS – SHUTDOWN
UNIT 2 ONLY

There are no specific No Significant Hazards Considerations for this Specification.

CTS 3/4.10.1, SHUTDOWN MARGIN

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

M01

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 40 gpm of 1900 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 40 gpm of 1900 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length CEA required either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

M01

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth, MTC, and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*.

ACTION:

a. With any full-length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length CEA required either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

* Operation in MODE 3 shall be limited to 6 consecutive hours.

DISCUSSION OF CHANGES
CTS 3/4.10.1, SHUTDOWN MARGIN

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.10.1 provides an exception to the SHUTDOWN MARGIN (SDM) requirements of CTS 3.1.1.1 in MODE 2 and Unit 2 CTS 3.10.1 provides an exception to the SHUTDOWN MARGIN requirements of CTS 3.1.1.1 in MODES 2 and 3 for the purpose of measurement of control element assembly (CEA) worth and shutdown margin, and moderator temperature coefficient (MTC) for Unit 2, provided the reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEAs. According to the Bases, this special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEA measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations. This changes the CTS by eliminating a special test exception.

This change is acceptable because SDM is maintained in MODE 2 by CEA alignment and insertion limits (CTS 3.1.3.1, 3.1.3.5, and 3.1.3.6), which are not suspended by CTS 3.10.1. In addition, verification of the actual versus predicted core reactivity condition is not required to be performed in MODE 3. ITS SR 3.1.2.1, Note 2 states that this Surveillance is not required to be performed prior to entry into MODE 2. As a result, the CTS 3.10.1 special test exception is not needed. Future PHYSICS TESTS, including CEA worth, SDM, and MTC measurements, may be performed under ITS 3.1.8, "Special Test Exceptions (STE) - MODES 1 and 2," or in accordance with the requirements of LCO 3.1.4, “Control Element Assembly (CEA) Alignment,” LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.6, “Regulating Control Element Assembly (CEA) Insertion Limits.” This change is designated as more restrictive because a special test exception allowance is being deleted.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.10.1, SHUTDOWN MARGIN

There are no specific No Significant Hazards Considerations for this Specification.

UNIT 1 CTS 3/4.10.5, Center CEA Misalignment
UNIT 2 CTS 3/4.10.4, Center CEA Misalignment

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

SPECIAL TEST EXCEPTIONS

M01

CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.5 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.5.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.5.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

SPECIAL TEST EXCEPTIONS

M01

3/4.10.4 CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

DISCUSSION OF CHANGES
UNIT 1 CTS 3/4.10.5, CENTER CEA MISALIGNMENT
UNIT 2 CTS 3/4.10.4, CENTER CEA MISALIGNMENT

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M01 Unit 1 Current Technical Specifications (CTS) 3/4.10.5 and Unit 2 CTS 3/4.10.5 provide an exception to CTS 3.1.3.3 and CTS 3.1.3.6 Specifications. This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient. The Improved Technical Specifications (ITS) do not contain this special test exception. This changes the CTS by eliminating a special test exception.

This change is acceptable because this type of PHYSICS TEST is no longer performed. Future PHYSICS TESTS will be performed under ITS 3.1.8, "Special Test Exceptions (STE) - MODES 1 and 2," or in accordance with the requirements of LCO 3.1.4, “Control Element Assembly (CEA) Alignment,” and LCO 3.1.6, “Regulating Control Element Assembly (CEA) Insertion Limits.” As a result, this CTS Special test exception is not needed. This change is designated as more restrictive because an exception to the CTS is being deleted.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
UNIT 1 CTS 3/4.10.5, CENTER CEA MISALIGNMENT
UNIT 2 CTS 3/4.10.4, CENTER CEA MISALIGNMENT

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 10

ISTS Not Adopted

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

1

3.1 REACTIVITY CONTROL SYSTEMS (Analog)

3.1.7 Special Test Exceptions (STE) - SHUTDOWN MARGIN (SDM) (Analog)

LCO 3.1.7 During performance of PHYSICS TESTS, the requirements of:

LCO 3.1.1, "SHUTDOWN MARGIN,"
LCO 3.1.5, "Shutdown Control Element Assembly Insertion Limits," and
LCO 3.1.6, "Regulating Control Element Assembly Insertion Limits,"

may be suspended for measurement of Control Element Assembly (CEA) worth, provided shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) is available for trip insertion.

APPLICABILITY: MODES 2 and 3 during PHYSICS TESTS.

NOTE
Operation in MODE 3 shall be limited to 6 consecutive hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any CEA not fully inserted and less than the above shutdown reactivity equivalent available for trip insertion. OR All CEAs inserted and the reactor subcritical by less than the above shutdown reactivity equivalent.	A.1 Initiate boration to restore required shutdown reactivity.	15 minutes

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify that the position of each CEA not fully inserted is within the acceptance criteria for available negative reactivity addition.	[2 hours OR In accordance with the Surveillance Frequency Control Program]
SR 3.1.7.2	NOTE Not required to be performed during initial power escalation following a refueling outage if SR 3.1.4.6 has been met. Verify that each CEA not fully inserted is capable of full insertion when tripped from at least the 50% withdrawn position.	Once within [7 days] prior to reducing SDM to less than the limits of LCO 3.1.1

JUSTIFICATION FOR DEVIATIONS
ISTS 3.1.7, SPECIAL TEST EXCEPTIONS (STE) - SHUTDOWN MARGIN (SDM) (ANALOG)

1. ISTS 3.1.7, "Special Test Exceptions (STE) - SHUTDOWN MARGIN (SDM) (Analog)," is not included in the St. Lucie Plant Unit 1 and Unit 2 ITS because the exception is not needed to perform any required startup or PHYSICS TESTS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

1

B 3.1 REACTIVITY CONTROL SYSTEMS (Analog)

B 3.1.7 Special Test Exception (STE) - SHUTDOWN MARGIN (SDM) (Analog)

BASES

BACKGROUND The primary purpose of the SHUTDOWN MARGIN (SDM) Special Test Exception (STE) is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are constructed to determine the control element assembly (CEA) worth.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed,

b. Validate the analytical models used in design and analysis,

c. Verify assumptions used for predicting plant response,

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design, and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 4).

1

BASES

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because adequate limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate (LHR) remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

a. LCO 3.1.1, "SHUTDOWN MARGIN (SDM)",

b. LCO 3.1.5, “Shutdown Control Element Assembly (CEA) Insertion Limits,” and

c. LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits."

Therefore, this LCO places limits on the minimum amount of CEA worth required to be available for reactivity control when CEA worth measurements are performed.

The individual LCOs cited above govern SDM CEA group height, insertion, and alignment. Additionally, the LCOs governing Reactor Coolant System (RCS) flow, reactor inlet temperature, and pressurizer pressure contribute to maintaining departure from nucleate boiling (DNB) parameter limits. The initial condition criteria for accidents sensitive to

APPLICABLE SAFETY ANALYSES (continued)

core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the loss of coolant accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 6). The criteria for the loss of forced reactor coolant flow accident are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

SRs are conducted as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these SRs allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

Requiring that shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) be available for trip insertion from the OPERABLE CEA provides a high degree of assurance that shutdown capability is maintained for the most challenging postulated accident, a stuck CEA. Since LCO 3.1.1 is suspended, however, there is not the same degree of assurance during this test that the reactor would always be shut down if the highest worth CEA was stuck out and calculational uncertainties or the estimated highest CEA worth was not as expected (the single failure criterion is not met). This situation is judged acceptable, however, because specified acceptable fuel damage limits are still met. The risk of experiencing a stuck CEA and subsequent criticality is reduced during this PHYSICS TEST exception by the requirements to determine CEA positions every 2 hours; by the trip of each CEA to be withdrawn 24 hours prior to suspending the SDM; and by ensuring that shutdown reactivity is available, equivalent to the reactivity worth of the estimated highest worth withdrawn CEA (Ref. 5).

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total planar radial peaking factor, total integrated radial peaking factor, T_q and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Special Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

1

BASES

LCO

This LCO provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. The STE is required to permit the periodic verification of the actual versus predicted worth of the regulating and shutdown CEAs. The SDM requirements of LCO 3.1.1, the shutdown CEA insertion limits of LCO 3.1.5, and the regulating CEA insertion limits of LCO 3.1.6 may be suspended.

APPLICABILITY

This LCO is applicable in MODES 2 and 3. Although CEA worth testing is conducted in MODE 2, sufficient negative reactivity is inserted during the performance of these tests to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the STE allows limited operation to 6 consecutive hours in MODE 3, as indicated by the Note, without having to borate to meet the SDM requirements of LCO 3.1.1.

ACTIONS

A.1

With any CEA not fully inserted and less than the minimum required reactivity equivalent available for insertion, or with all CEAs inserted and the reactor subcritical by less than the reactivity equivalent of the highest worth CEA, restoration of the minimum SDM requirements must be accomplished by increasing the RCS boron concentration. The required Completion Time of 15 minutes for initiating boration allows the operator sufficient time to align the valves and start the boric acid pumps and is consistent with the Completion Time of LCO 3.1.1.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification of the position of each partially or fully withdrawn full length or part length CEA is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. [A 2 hour Frequency is sufficient for the operator to verify that each CEA position is within the acceptance criteria.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

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1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.2

Prior demonstration that each CEA to be withdrawn from the core during PHYSICS TESTS is capable of full insertion, when tripped from at least a 50% withdrawn position, ensures that the CEA will insert on a trip signal. The Frequency ensures that the CEAs are OPERABLE prior to reducing SDM to less than the limits of LCO 3.1.1.

The SR is modified by a Note which allows the SR to not be performed during initial power escalation following a refueling outage if SR 3.1.4.6 has been met during that refueling outage. This allows the CEA drop time test, which also proves the CEAs are trippable, to be credited for this SR.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August 1978.

4. ANSI/ANS-19.6.1-1985, December 13, 1985.

5. FSAR, Chapter [14].

6. 10 CFR 50.46.

7. FSAR, Chapter [15].

JUSTIFICATION FOR DEVIATIONS
ISTS 3.1.7 BASES, SPECIAL TEST EXCEPTIONS (STE) - SHUTDOWN MARGIN (SDM) (ANALOG)

1. ISTS 3.1.7, "Special Test Exceptions (STE) - SHUTDOWN MARGIN (SDM) (Analog)," is not included in the St. Lucie Plant (PSL) Unit 1 and Unit 2 ITS because the exception is not needed to perform any required startup or PHYSICS TESTS. Therefore, the Bases associated with ISTS 3.1.7 is not included in the PSL Unit 1 and Unit 2 ITS Bases.

ENCLOSURE 2

VOLUME 7

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.2
POWER DISTRIBUTION LIMITS

Revision 1

LIST OF ATTACHMENTS

1. ITS 3.2.1, LINEAR HEAT RATE (LHR)
2. ITS 3.2.2, TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T
3. ITS 3.2.3, AZIMUTHAL POWER TILT (Tq)
4. ITS 3.2.4, AXIAL SHAPE INDEX (ASI)
5. ISTS NOT ADOPTED

ATTACHMENT 1

ITS 3.2.1, LINEAR HEAT RATE (LHR)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

~~3/4.2~~ POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE (LHR)

LIMITING CONDITION FOR OPERATION

LCO 3.2.1

3.2.1 ~~The linear heat rate~~ LHR shall not exceed the limits specified in the COLR.

Applicability

APPLICABILITY: MODE 1.

ACTION:

ACTION A

~~With the linear heat rate~~ LHR, as determined by the Excore Detector Monitoring system, exceed~~ing its~~ limits, as indicated by four or more coincident incore channels or LHR, as determined by the Incore Detector Monitoring system, by the ~~AXIAL SHAPE INDEX~~ ASI outside ~~of~~ s the specified in the COLR the power dependent control limits ~~of~~ specified in the COLR ~~Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:~~ L01

Required Action A.1

a. Restore ~~the linear heat rate~~ LHR to within ~~its~~ limits within one hour, or

Required Action B.1

b. Be in ~~HOT STANDBY~~ MODE 2 within the next 6 hours. L02

SURVEILLANCE REQUIREMENTS

~~4.2.1.1 The provisions of Specification 4.0.4 are not applicable.~~ M01

SR Note

~~4.2.1.2 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with~~ either the excore detector monitoring system or ~~with~~ the incore detector monitoring system. shall be used to determine LHR

SR 3.2.1.1 Note

~~4.2.1.3 Excore Detector Monitoring System~~ The excore detector monitoring system ~~may be~~ is being used ~~for monitoring the linear heat rate by:~~ to determine LHR. Only required to be met when

a. Verifying in accordance with the Surveillance Frequency Control Program that the full length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6. See ITS 3.1.6

SR 3.2.1.1

b. Verify~~ing~~ in accordance with the Surveillance Frequency Control Program ~~that the AXIAL SHAPE INDEX~~ alarm setpoints are ~~adjusted to~~ within the limits ~~shown on~~ specified in the COLR ~~Figure 3.2-2~~. ASI

A01

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (continued)

c. ~~Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of COLR Figure 3.2-2, where 100 percent of maximum allowable power represents the maximum THERMAL POWER allowed by the following expression:~~

~~**M x N**~~

~~where:~~

~~1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.~~

~~2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F_r^T curve of COLR Figure 3.2-3.~~

LA01

SR 3.2.1.2
Note 1 and 2
SR 3.2.1.3
Note 1 and 2

4.2.1.4 ~~Incore Detector Monitoring System~~ # ~~shall~~ The incore detector monitoring system ~~may be~~ used ~~for monitoring the linear heat rate by verifying that the incore detector~~ to determine LHR. Local Power Density alarms~~:~~

is being

1. Only required to be met when

Verify incore detector

2. Not required to be performed below 20% RTP.

L03

SR 3.2.1.2

a. ~~Are adjusted to~~ satisfy the requirements of the core power distribution map which shall be updated in accordance with the Surveillance Frequency Control Program ~~in MODE 1~~.

Verify incore detector local power density

SR 3.2.1.3

b. ~~Have their~~ alarm setpoint~~s~~ ~~adjusted to~~ are less than or equal to the limits ~~shown on~~ specified in the COLR ~~Figure 3.2-1~~.

in accordance with the Surveillance Frequency Control Program

M02

~~# If the incore system become inoperable, reduce power to M x N within 4 hours and monitor linear heat rate in accordance with Specification 4.2.1.3.~~

LA01

Pages 3/4 2-4 (Amendment 106), 3/4 2-5 (Amendment 63), and 3/4 2-6 through 3/4 2-8 (Amendment 109) have been deleted from the Technical Specifications. The next page is 3/4 2-9.

A01

A01

~~3/4.2~~ POWER DISTRIBUTION LIMITS

~~3/4 2.1~~ LINEAR HEAT RATE (LHR)

LIMITING CONDITION FOR OPERATION

LCO 3.2.1 3.2.1 ~~The linear heat rate~~ (LHR) shall not exceed the limits specified in the COLR.

Applicability **APPLICABILITY:** MODE 1.

ACTION:

ACTION A ~~With the linear heat rate~~ exceed~~ing its~~ limits, as indicated by four or more coincident incore channels or by the ~~AXIAL SHAPE INDEX~~ outside ~~of~~ the power dependent control limits ~~of~~ COLR ~~Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:~~ L01

Required Action A.1 a. Restore ~~the linear heat rate~~ to within ~~its~~ limits within 1 hour, or

Required Action B.1 b. Be in ~~at least HOT STANDBY~~ MODE 2 within the next 6 hours. L02

SURVEILLANCE REQUIREMENTS

~~4.2.1.1~~ ~~The provisions of Specification 4.0.4 are not applicable.~~ M01

SR Note ~~4.2.1.2~~ ~~The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with~~ either the excore detector monitoring system or ~~with~~ the incore detector monitoring system shall be used to determine LHR

SR 3.2.1.1 Note ~~4.2.1.3~~ ~~Excore Detector Monitoring System –~~ The excore detector monitoring system ~~may be~~ used ~~for monitoring the linear heat rate by:~~ to determine LHR.

is being

a. Verifying in accordance with the Surveillance Frequency Control Program that the full-length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6. See ITS 3.1.6

SR 3.2.1.1 ~~b.~~ Verify~~ing~~ in accordance with the Surveillance Frequency Control Program ~~that the AXIAL SHAPE INDEX~~ alarm setpoints are ~~adjusted to~~ within the limits ~~shown on~~ COLR ~~Figure~~ 3.2-2.

specified in the ASI

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

~~c. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of COLR Figure 3.2-2, where 100% of maximum allowable power represents the maximum THERMAL POWER allowed by the following expression:~~

LA01

~~**M x N**~~

~~where:~~

~~1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.~~

~~2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F_r^T curve of COLR Figure 3.2-3.~~

A01

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.2.1.2 Note 1 and 2
SR 3.2.1.3 Note 1 and 2

1. Only required to be met when

~~4.2.1.4 Incore Detector Monitoring System#~~ The incore detector monitoring system ~~may be~~ used ~~for monitoring the linear rate by verifying that the incore detector~~ Local Power Density alarms:

is being

to determine LHR.

Verify incore detector

2. Not required to be performed below 20% RTP.

L03

SR 3.2.1.2

~~a.~~ ~~Are adjusted to~~ satisfy the requirements of the core power distribution map which shall be updated in accordance with the Surveillance Frequency Control Program in MODE 1.

Verify incore detector local power density

s

are

SR 3.2.1.3

~~b.~~ ~~Have their~~ alarm setpoint ~~adjusted to~~ less than or equal to the limits ~~shown on~~ COLR ~~Figure 3.2-1~~.

specified in the

in accordance with the Surveillance Frequency Control Program

M02

~~# If incore system becomes inoperable, reduce power to M x N within 4 hours and monitor linear heat rate in accordance with Specification 4.2.1.3.~~

LA01

Pages 3/4 2-4 (Amendment 42), 3/4 2-5 (Amendment 8), and 3/4 2-6 (Amendment 17) have been deleted from the Technical Specifications. The next page is 3/4 2-7.

A01

~~POWER DISTRIBUTION LIMITS~~

A01

~~DELETED~~

A01

~~POWER DISTRIBUTION LIMITS~~

A01

~~DELETED~~

A01

DISCUSSION OF CHANGES
ITS 3.2.1, LINEAR HEAT RATE (LHR)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 4.2.1.1 states that the provisions of Specification 4.0.4 are not applicable, and thereby provides an allowance for entering the next higher MODE of Applicability when the Surveillance is not met. ITS 3.2.1 does not provide a Surveillance Note that states that the provisions of LCO 4.0.4 are not applicable. LCO 4.0.4 states that entry into a MODE or other specified condition in the Applicability of a Limiting Condition for Operation (LCO) shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by Surveillance Requirement 4.0.3. LCO 4.0.3 provides the provisions for a missed surveillance and does not apply to a surveillance known to not be met within its specified Frequency prior to entering the Mode of Applicability.

The purpose of CTS 4.2.2.1 is to provide an allowance for entering the MODE of applicability when any Surveillance is not met. This change is designated as more restrictive because the CTS 4.0.4 MODE change allowance is deleted and entry into MODE 1 shall only be made when the LCO's surveillances have been met within their specified Frequency. This changes CTS by allowing entry into the MODE of Applicability by only deferring the performance of the Surveillance Requirements instead of deferring compliance with the LCO.

ITS SR 3.2.1.2 and SR 3.2.1.3 each contain a note that states "Not required to be performed below 20% RTP. ITS SR 3.2.1.2 and SR 3.2.1.3 Surveillance Requirements Note will provide an allowance whereby, Surveillance performance is not required prior to entering MODE 1. This allows establishing the conditions at which the Surveillances can be performed. See DOC L03 discussion of change for ITS SR 3.2.1.2 Note and SR 3.2.1.3 Note.

M02 CTS 4.2.1.4.b does not explicitly state how often the incore detector low power density alarm setpoint is adjusted to less than or equal to the limits shown on COLR Figure 3.2.1. CTS 4.2.1.4.b is changed to include a Surveillance Frequency in accordance with the Surveillance Frequency Control Program (SFCP) consistent with the Frequency to update the core distribution map as specified in CTS 4.2.1.4.a. The Frequency of ITS 3.2.1.3 is "in accordance with the Surveillance Frequency Control Program."

PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Performance of SR 3.2.1.3 (CTS 4.2.1.4.b) verifies, in part, that the Incore Detector Monitoring System can accurately monitor LHR and is performed following the update of the core flux map, which is performed at a frequency specified in the SFCP. The change to the Frequency of CTS 4.2.1.4.a is discussed in FPL (PSL Unit 1 and Unit 2) "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program" (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066). The initial frequency established in accordance with the SFCP will be 31 days consistent with the frequency established for ISTS SR 3.2.1.3 and considers the historical testing frequency of the reactor monitoring system.

This change is designated as more restrictive because a Surveillance Frequency has been added to ensure the incore detector local power density alarm setpoints are periodically verified to be within the limits of the COLR.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)*

CTS 4.2.1.4 footnote # states that if the incore detector monitoring system becomes inoperable, reduce power to M x N within 4 hours and monitor linear heat rate in accordance with Specification 4.2.1.3. CTS 4.2.1.3.c provides the determination of M x N and directs which figures in the COLR to use for LHR verifications using the excore detection system. ITS 3.2.1 does not include this requirement. This changes the CTS by relocating the action to take when the incore detector monitoring system is inoperable to the Technical Requirements Manual (TRM).

The incore detector monitoring system monitors LHR. However, there are no requirements in CTS for this indication-only instrumentation system to be OPERABLE. Therefore, it is unnecessary to provide actions in the Technical Specifications that must be performed when this non-TS system is not functional. The removal of action requirements for indication-only instrumentation and alarms from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The requirement to determine LHR and actions to perform when LHR is not within limits remains in

the ITS. Either the incore detector monitoring system or the excore detector monitoring system is used to monitor LHR.

Normally, LHR is verified using the incore detector local power density alarm. The setpoint is maintained less than or equal to the COLR Figure 3.2-1 limits. When the incore detector monitoring system is discovered to be non-functional, the ASI alarm associated with the excore detector monitoring system is used to determine LHR. The ASI alarm setpoint is maintained within the COLR Figure 3.2-2 limits. A power reduction may be required to adjust the ASI alarm setpoint to correspond to the COLR figure limits, however, a power reduction may not be necessary based on plant conditions. If a power reduction is necessary to adjust the ASI alarm setpoints to within COLR limits, the power reduction is accomplished using normal plant operating procedures.

This change is acceptable because inoperability of the incore detector monitoring system alone does not indicate that LHR has exceeded the limit and the removed information will be adequately controlled in the TRM. Changes to the TRM are made under 10 CFR 50.59, thereby ensuring changes are properly evaluated. This change is designated as a less restrictive removal of detail change because action requirements for non-TS indication-only instrumentation and alarms is being removed from the Technical Specifications. R1

LESS RESTRICTIVE CHANGES

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.1 Action states that within 15 minutes initiate corrective action to reduce the linear heat rate (LHR) to within limits. ITS 3.2.1 Required Action A.1 and associated Completion Time states that LHR be restored to within limits with a Completion Time of 1 hour. This changes the CTS by not requiring that within 15 minutes corrective action be taken to reduce the LHR to within limits, rather only requiring that the LHR be restored to within limits within 1 hour, consistent with the ITS 3.2.1 Completion Time. Requiring corrective action be taken within 15 minutes does not necessarily restore LHR within limits within 1 hour.

The purpose of CTS 3.2.1 is to maintain LHR within the limits specified in the COLR. When the LCO is not met, both CTS 3.2.1 and ITS 3.2.1 require that the LHR be restored to within limits within 1 hour. With the LHR exceeding its limit, fuel damage could occur following an accident. In this Condition, prompt action must be taken to restore the LHR to within the specified limits. One hour to restore the LHR to within its specified limits is reasonable and ensures that the core does not continue to operate in this Condition. The 1 hour Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions. This change is acceptable because the CTS 3.2.1 Completion Time and ITS 3.2.1 Completion Time are both 1 hour.

This change is designated as less restrictive because the CTS requirement to within 15 minutes initiate corrective action to reduce the linear heat rate (LHR) to within limits is deleted.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.2.1 Action b. states that with the linear heat rate (LHR) not within limits within 1 hour, to be in HOT STANDBY within the next 6 hours. ITS 3.2.1 Action B states that with the LHR not within limits within 1 hour, then be in MODE 2 with a Completion Time of 6 hours. This changes the CTS by only requiring entry into MODE 2 within 6 hours. Once in MODE 2, LCO 3.0.2 allows the LCO to be exited since the Applicability is MODE 1, and completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The purpose of CTS 3.2.1 Action b. and ITS 3.2.1 Action B is to reduce THERMAL POWER. Exiting the Mode of Applicability provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.2.1.1 states that the provisions of Specification 4.0.4 are not applicable, and thereby provides an allowance for entering the next higher MODE of Applicability when the Surveillance is not met. ITS 3.2.1 does not provide a Surveillance Note that states that the provisions of LCO 4.0.4 are not applicable. See DOC M01. ITS SR 3.2.1.2 and SR 3.2.1.3 each contain a note that states “Not required to be performed below 20% RTP.” ITS SR 3.2.1.2 and SR 3.2.1.3 Surveillance Requirements Note will provide an allowance whereby, Surveillance performance is not required prior to entering MODE 1. This allows establishing the conditions at which the Surveillances can be performed. CTS 4.0.1. states that Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual LCOs unless otherwise stated in an individual Surveillance Requirement. Similarly, ITS SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. This changes the CTS by providing a Surveillance Note which states “Not required to be performed below 20% RTP.”

The purpose of CTS 4.2.1.1 is to provide a note that states that the provisions of Specification 4.0.4 are not applicable, and thereby provides an allowance for entering the next higher MODE of Applicability when the Surveillance is not met. ITS 3.2.1 does not provide a Surveillance Note that states that the provisions of LCO 4.0.4 are not applicable.

This change is acceptable because the note added to ITS SR 3.2.1.2 and SR 3.2.1.3 will provide an allowance whereby, Surveillance performance is not

met prior to entering MODE 1. This allows establishing the conditions at which the Surveillances can be performed. This change is designated as less restrictive because Surveillances will be performed after entering MODE 1 rather than being performed prior to entry into MODE 1.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

4

LCO 3.2.1 3.2.1 Linear Heat Rate (LHR) ~~(Analog)~~

LCO 3.2.1 LCO 3.2.1 LHR shall not exceed the limits specified in the COLR.

Applicability APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A A. LHR, as determined by the Incore Detector Monitoring System, exceeds the limits specified in the COLR, as indicated by four or more coincident incore channels. OR LHR, as determined by the Excore Detector Monitoring System, exceeds the limits as indicated by the ASI outside the power dependent control limits specified in the COLR.	A.1 Restore LHR to within limits.	1 hour
ACTION B B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

4

SURVEILLANCE REQUIREMENTS

----------NOTE----------

SR 4.2.1.2 Either the Excore Detector Monitoring System or the Incore Detector Monitoring System shall be used to determine LHR.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	----------NOTE---------- Only required to be met when the Excore Detector Monitoring System is being used to determine LHR. ----------	
SR 4.2.1.3		
SR 4.2.1.3.b	Verify ASI alarm setpoints are within the limits specified in the COLR.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 4.2.1.4.a SR 3.2.1.2	----------NOTES---------- 1. Only required to be met when the Incore Detector Monitoring System is being used to determine LHR. 2. Not required to be performed below 20% RTP. ----------	
	Verify incore detector local power density alarms satisfy the requirements of the core power distribution map~~, which shall be updated at least once per 31 days of accumulated operation in MODE 1~~.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

2

2

2

3

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 4.2.1.4.b	SR 3.2.1.3	----------NOTES---------- 1. Only required to be met when the Incore Detector Monitoring System is being used to determine LHR. 2. Not required to be performed below 20% RTP. ----------
	Verify incore detector local power density alarm setpoints are less than or equal to the limits specified in the COLR.	~~[31 days~~ 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

4

LCO 3.2.1 3.2.1 Linear Heat Rate (LHR) ~~(Analog)~~

LCO 3.2.1 LCO 3.2.1 LHR shall not exceed the limits specified in the COLR.

Applicability APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A A. LHR, as determined by the Incore Detector Monitoring System, exceeds the limits specified in the COLR, as indicated by four or more coincident incore channels. OR LHR, as determined by the Excore Detector Monitoring System, exceeds the limits as indicated by the ASI outside the power dependent control limits specified in the COLR.	A.1 Restore LHR to within limits.	1 hour
ACTION B B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

4

SURVEILLANCE REQUIREMENTS

----------NOTE----------

SR 4.2.1.2 Either the Excore Detector Monitoring System or the Incore Detector Monitoring System shall be used to determine LHR.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	----------NOTE---------- Only required to be met when the Excore Detector Monitoring System is being used to determine LHR. ----------	
SR 4.2.1.3		
SR 4.2.1.3.b	Verify ASI alarm setpoints are within the limits specified in the COLR.	~~[31 days~~ ~~OR~~ 2 In accordance with the Surveillance Frequency Control Program~~]~~ 2
SR 4.2.1.4.a SR 3.2.1.2	----------NOTES---------- 1. Only required to be met when the Incore Detector Monitoring System is being used to determine LHR. 2. Not required to be performed below 20% RTP. ----------	
	Verify incore detector local power density alarms satisfy the requirements of the core power distribution map~~, which shall be updated at least once per 31 days of accumulated operation in MODE 1~~.	~~[31 days~~ 2 ~~OR~~ 3 In accordance with the Surveillance Frequency Control Program~~]~~

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 4.2.1.4.b	SR 3.2.1.3	----------NOTES---------- 1. Only required to be met when the Incore Detector Monitoring System is being used to determine LHR. 2. Not required to be performed below 20% RTP. ----------
	Verify incore detector local power density alarm setpoints are less than or equal to the limits specified in the COLR.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.1, LINEAR HEAT RATE (LHR)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS SR 3.2.1.2 statement "which shall be updated at least once per 31 days of accumulated operation in MODE 1" is deleted. ITS SR 3.2.1.2, including update of the core power distribution map, is in accordance with the Surveillance Frequency Control Program (SFCP). The SFCP Frequency is at least once per 31 days.

4. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

5

B 3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

5

B 3.2.1 Linear Heat Rate (LHR) ~~(Analog)~~

BASES

(Linear Heat Rate (LHR))

1

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution,

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution, and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of ~~satisfies~~ this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Ref~~s~~. 1 ~~and 2~~), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

1

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on linear heat rate (LHR) and departure from nucleate boiling (DNB).

1

BASES

BACKGROUND (continued)

The limits on LHR, ~~Total Planar Radial Peaking Factor (F^T_{XY}),~~ Total Integrated Radial Peaking Factor (F^T_r), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR is within its limits. The Excore Detector Monitoring System performs this function by continuously monitoring ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing ASI limits, the following ~~assumptions~~ are ~~made~~:

a. The CEA insertion limits of LCO 3.1.5, "Shutdown CEA Insertion Limits," and LCO 3.1.6, "Regulating CEA Insertion Limits," are satisfied,

b. The T_q restrictions of LCO 3.2.~~4~~ are satisfied, and

c. ~~F^T_{xy}~~ is within the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors and alarms ~~that have been~~ established for the individual incore detector segments~~,~~ ensuring that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances~~, set in conservative directions, for:~~

~~a. A measurement calculational uncertainty factor of 1.062,~~

~~b. An engineering uncertainty factor of 1.03,~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion, and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02.~~

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and AOOs (Condition 2) (Ref. ~~3~~, GDC 10). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. ~~4~~),

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. ~~3~~, GDC 10),

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 2~~8~~0 cal/gm (Ref. [ ]), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. ~~3~~, GDC 26).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. ~~4~~). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by accident analyses (Ref. 1), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. ~~4~~). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI~~,~~ ~~F^T_{XY}~~, F^T_r ~~, and T_q~~ limits specified in the COLR. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

5

BASES

APPLICABLE SAFETY ANALYSES (continued)

reactor

1

Fuel cladding damage does not normally occur while the ~~unit~~ is operating at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. ~~The~~ This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR.

1

The LHR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB ratio operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on the LHR ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because ~~there~~ is not sufficient THERMAL POWER to require a limit on the core power distribution.

1

ACTIONS

A.1

With the LHR exceeding its limit, excessive fuel damage could occur following an accident. In this Condition, prompt action must be taken to restore the LHR to within the specified limits. One hour to restore the LHR to within its specified limits is reasonable and ensures that the core does not continue to operate in this Condition. The 1 hour Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

B.1

If the LHR cannot be returned to within its specified limits, THERMAL POWER must be reduced. The change to MODE 2 provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power MODE 1 conditions in an orderly manner and without challenging plant systems.

1

5

BASES

SURVEILLANCE REQUIREMENTS

A Note was added to the SRs to require LHR to be determined by either the Excore Detector Monitoring System or the Incore Detector Monitoring System.

SR 3.2.1.1

Performance of this SR verifies that the Excore Detector Monitoring System can accurately monitor the LHR. Therefore, this SR is only applicable when the Excore Detector Monitoring System is being used to determine the LHR. ~~[The 31 day Frequency is appropriate for this SR because it is consistent with the requirements of SR 3.3.1.3 for calibration of the excore detectors using the incore detectors.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

The SR is modified by a Note that states that the SR is only required to be met when the Excore Detection Monitoring System is being used to determine LHR. The reason for the Note is that the excore detectors input neutron flux information into the ASI calculation.

SR 3.2.1.2 and SR 3.2.1.3

Continuous monitoring of the LHR is provided by the Incore Detector Monitoring System and the Excore Detector Monitoring System. Either of these two core power distribution monitoring systems provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its specified limits.

Performance of these SRs verifies that the Incore Detector Monitoring System can accurately monitor LHR. Therefore, they are only applicable when the Incore Detector Monitoring System is being used to determine the LHR.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[A 31 day Frequency is consistent with the historical testing frequency of the reactor monitoring system.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

The SRs are modified by two Notes. Note 1 allows the SRs to be met only when the Incore Detector Monitoring System is being used to determine LHR. Note 2 states that the SRs are not required to be performed when THERMAL POWER is < 20% RTP. The accuracy of the neutron flux information from the incore detectors is not reliable at ~~THERMAL POWER < 20% RTP~~.

REFERENCES

1. FSAR, Chapter ~~[~~15~~]~~.
2. ~~FSAR, Chapter [6].~~ NUREG 0800, Section 4.2, Appendix B
3. Technical Requirements Manual
4. ~~3.~~ 10 CFR 50, Appendix A.
5. ~~4.~~ 10 CFR 50.46.

low power levels . Note 2 allows entry into and operation in MODE 1 prior to performing the SRs in order to establish a power level that provides more accurate neutron flux information . Consistent with the requirements of SR 3.0.1 and SR 3.0.2, as specified in Example 1.4-5 of Section 1.4, "Frequency," the Surveillances must be performed within the Frequency requirements of the Surveillance Frequency Control Program prior to exceeding 20% RTP.

5

B 3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

5

B 3.2.1 Linear Heat Rate (LHR) ~~(Analog)~~

BASES

(Linear Heat Rate (LHR))

1

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution,

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution, and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of ~~satisfies~~ this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Ref~~s~~. 1 ~~and 2~~), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

1

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on linear heat rate (LHR) and departure from nucleate boiling (DNB).

1

BASES

BACKGROUND (continued)

The limits on LHR, ~~Total Planar Radial Peaking Factor (F^T_{XY}),~~ Total Integrated Radial Peaking Factor (F^T_r), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR is within its limits. The Excore Detector Monitoring System performs this function by continuously monitoring ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing ASI limits, the following ~~assumptions~~ are ~~made~~:

a. The CEA insertion limits of LCO 3.1.5, "Shutdown CEA Insertion Limits," and LCO 3.1.6, "Regulating CEA Insertion Limits," are satisfied,

b. The T_q restrictions of LCO 3.2.~~4~~ are satisfied, and

c. ~~F^T_{xy}~~ is within the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors and alarms ~~that have been~~ established for the individual incore detector segments~~,~~ ensuring that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances~~, set in conservative directions, for:~~

~~a. A measurement calculational uncertainty factor of 1.062,~~

~~b. An engineering uncertainty factor of 1.03,~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion, and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02.~~

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and AOOs (Condition 2) (Ref. ~~3~~, GDC 10). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. ~~4~~),

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. ~~3~~, GDC 10),

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 2~~8~~0 cal/gm (Ref. [ ]), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. ~~3~~, GDC 26).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. ~~4~~). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by accident analyses (Ref. 1), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. ~~4~~). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI~~,~~ ~~F^T_{XY},~~ F^T_r ~~, and T_q~~ limits specified in the COLR. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not normally occur while the ~~unit~~ reactor is operating at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. ~~The~~ This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR.

The LHR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB ratio operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on the LHR ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because ~~there~~ is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1

With the LHR exceeding its limit, excessive fuel damage could occur following an accident. In this Condition, prompt action must be taken to restore the LHR to within the specified limits. One hour to restore the LHR to within its specified limits is reasonable and ensures that the core does not continue to operate in this Condition. The 1 hour Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

B.1

If the LHR cannot be returned to within its specified limits, THERMAL POWER must be reduced. The change to MODE 2 provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power MODE 1 conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

A Note was added to the SRs to require LHR to be determined by either the Excore Detector Monitoring System or the Incore Detector Monitoring System.

SR 3.2.1.1

Performance of this SR verifies that the Excore Detector Monitoring System can accurately monitor the LHR. Therefore, this SR is only applicable when the Excore Detector Monitoring System is being used to determine the LHR. ~~[The 31 day Frequency is appropriate for this SR because it is consistent with the requirements of SR 3.3.1.3 for calibration of the excore detectors using the incore detectors.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

The SR is modified by a Note that states that the SR is only required to be met when the Excore Detection Monitoring System is being used to determine LHR. The reason for the Note is that the excore detectors input neutron flux information into the ASI calculation.

SR 3.2.1.2 and SR 3.2.1.3

Continuous monitoring of the LHR is provided by the Incore Detector Monitoring System and the Excore Detector Monitoring System. Either of these two core power distribution monitoring systems provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its specified limits.

Performance of these SRs verifies that the Incore Detector Monitoring System can accurately monitor LHR. Therefore, they are only applicable when the Incore Detector Monitoring System is being used to determine the LHR.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[A 31 day Frequency is consistent with the historical testing frequency of the reactor monitoring system.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

The SRs are modified by two Notes. Note 1 allows the SRs to be met only when the Incore Detector Monitoring System is being used to determine LHR. Note 2 states that the SRs are not required to be performed when THERMAL POWER is < 20% RTP. The accuracy of the neutron flux information from the incore detectors is not reliable at ~~THERMAL POWER < 20% RTP~~.

REFERENCES

1. FSAR, Chapter ~~[~~15~~]~~.
2. ~~FSAR, Chapter [6]~~.
3. 10 CFR 50, Appendix A.
4. 10 CFR 50.46.

low power levels . Note 2 allows entry into and operation in MODE 1 prior to performing the SRs in order to establish a power level that provides more accurate neutron flux information . Consistent with the requirements of SR 3.0.1 and SR 3.0.2, as specified in Example 1.4-5 of Section 1.4, "Frequency," the Surveillances must be performed within the Frequency requirements of the Surveillance Frequency Control Program prior to exceeding 20% RTP.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.1, BASES, LINEAR HEAT RATE (LHR)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS contains tolerances to be applied to the incore detector alarm setpoints. PSL Unit 1 UFSAR, Section 13.8, Licensee-Controlled Technical Specification Requirements, and PSL Unit 2 UFSAR, Section 13.7, Licensee-Controlled Technical Specification Requirements, each contain these tolerances to be applied to the incore detector alarm setpoints. Additionally, these tolerances will be provided in the Technical Requirements Manual. Therefore, the tolerances to be applied to the incore detector alarm setpoints are deleted.

4. The ISTS includes ISTS 3.2.1, Linear Heat Rate (LHR), ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}), ISTS 3.2.3, Total Integrated Radial Peaking Factor (F_{r}^{T}), ISTS 3.2.4, Azimuthal Power Tilt (T_q), and ISTS 3.2.5, Axial Shape Index (ASI). CTS do not include a Specification for ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}). The CTS and ISTS are renumbered. CTS 3.2.3, Total Integrated Radial Peaking Factor (F_{r}^{T}), CTS 3.2.4, Azimuthal Power Tilt (T_q), and CTS 3.2.5, Axial Shape Index (ASI), are renumbered as ITS 3.2.2, ITS 3.2.3, and ITS 3.2.4, respectively.

5. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

6. Changes made to clarify the SR Frequency and Note 2 consistent with the generic Surveillance requirements specified in ISTS Section 3.0, “LCO and SR Applicability,” as specified in ISTS Section 1.4, “Frequency.”

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.1, LINEAR HEAT RATE (LHR)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

ITS 3.2.2, TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

POWER DISTRIBUTION LIMITS

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

LCO 3.2.2 3.2.~~3~~ 2 The calculated value of F_r^T shall be within the limits specified in the COLR.

Applicability APPLICABILITY: MODE 1~~*~~.

A03

ACTION:

ACTION A Note: Required Actions shall be completed if this Condition is entered.

ACTION A ~~With~~ F_r^T not within limit~~s~~, within 6 hours ~~either~~:

LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits," as specified in the COLR

L02

L01

Required Action B.1 ~~a.~~ Be in MODE 2 ~~at least HOT STANDBY, or~~

Required Action A.1 ~~b.~~ Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within ~~the~~ limits specified in the ~~of~~ COLR ~~Figure 3.2-3~~ and withdraw the ~~full length~~ control element assemblies ( ) CEAs to or beyond the Long Term Steady State Insertion Limits of ~~Specification 3.1.3.6~~. ~~The THERMAL POWER limit determined from COLR Figure 3.2-3 shall then be used to~~

Required Action A.2

Required Action A.3 establish a revised upper THERMAL POWER ~~level~~ limit as specified in the ~~on~~ COLR ~~Figure 3.2-4 (truncate Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by COLR Figure 3.2-3) and subsequent operation shall be maintained within the reduced acceptable operation region of COLR Figure 3.2-4.~~

LA01

LA01

SURVEILLANCE REQUIREMENTS

M01

~~4.2.3.1 The provisions of Specification 4.0.4 are not applicable.~~

SR 3.2.2.1 Note ~~4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r (1 + T_q)$ when F_r is calculated with a non-full core power distribution analysis code and shall be calculated as $F_r^T = F_r$ when calculations are performed with a full core power distribution analysis code.~~ F_r^T ~~shall be determined to be within its limit at the following intervals~~.

LA02

SR 3.2.2.1 Verify the value of

SR 3.2.2.1 Frequency > 70% RTP

a. Prior to operation ~~above 70 percent of RATED THERMAL POWER~~ after each fuel loading.

SR 3.2.2.1 Frequency

b. In accordance with the Surveillance Frequency Control Program ~~in MODE 1~~, and

~~c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.03.~~

A02

SR 3.2.2.1 and SR 3.2.2.3 shall be completed each time SR 3.2.2.1 is required. F^T_r shall be determined using the incore detectors to obtain a power distribution map with all CEAs at or above the long term steady state insertion limit as specified in the COLR.

~~* See Special Test Exception 3.10.2~~

A03

A01

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.2.2.2 Verify the value of SR 3.2.2.2 and SR 3.2.2.3 completed SR 3.2.2.1

SR 3.2.2.1 Note 4.2.3.3 ~~determined~~ F_r shall be ~~determined~~ each time ~~a calculation of~~ F_r^T is required ~~by~~ using the incore detectors to obtain a power distribution map with all ~~full length~~ CEAs at or above the Long Term Steady State Insertion Limit ~~for the existing Reactor Coolant Pump combination.~~ as specified in the COLR

F_r^T shall be determined

SR 3.2.2.3 Verify the value of SR 3.2.2.2 and SR 3.2.2.3 completed SR 3.2.2.1 required

SR 3.2.2.1 Note 4.2.3.4 ~~determined~~ T_q shall be ~~determined~~ each time ~~a calculation of~~ F_r^T is ~~made~~ ~~using a non-full core power distribution analysis code. The value of T_q used to determine F_r^T in this case shall be the measured value of T_q~~.

LA02

SR 3.2.2.2 Frequency
SR 3.2.2.3 Frequency

In accordance with the Frequency requirements of SR 3.2.2.1.

A01

POWER DISTRIBUTION LIMITS

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

LCO 3.2.2 3.2.3 The calculated value of F_r^T shall be within the limits specified in COLR.

Applicability APPLICABILITY: MODE 1*.

A03

ACTION:

ACTION A Note: Required Actions shall be completed if this Condition is entered.

ACTION A: With F_r^T not within limits, within 6 hours either:

MODE 2

Required Action B.1 a. Be in at least HOT STANDBY, or

specified in the

Required Action A.1 / Required Action A.2 b. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of COLR Figure 3.2-3 and withdraw the full-length control element assemblies ( ) to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6. The THERMAL POWER limit determined from COLR Figure 3.2-3 shall then be used to establish a revised upper THERMAL POWER level limit on COLR Figure 3.2-4 (truncate COLR Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by COLR Figure 3.2-3) and subsequent operation shall be maintained within the reduced acceptable operation region of COLR Figure 3.2-4.

LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits," as specified in the COLR

L02

L01

Required Action A.3 as specified in the

LA01

LA01

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

M01

SR 3.2.2.1 Note 4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r (1 + T_q)$ when F_r is calculated with a non-full core power distribution analysis code and shall be calculated as $F_r^T = F_r$ when calculations are performed with a full core power distribution analysis code. F_r^T shall be determined to be within its limit at the following intervals:

LA02

Verify the value of

SR 3.2.2.1

> 70% RTP

SR 3.2.2.1 Frequency a. Prior to operation above 70% of RATED THERMAL POWER after each fuel loading,

SR 3.2.2.1 Frequency b. In accordance with the Surveillance Frequency Control Program in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.03.

A02

SR 3.2.2.1 and SR 3.2.2.3 shall be completed each time SR 3.2.2.1 is required. F^T_r shall be determined using the incore detectors to obtain a power distribution map with all CEAs at or above the long term steady state insertion limit as specified in the COLR.

* See Special Test Exception 3.10.2

A03

A01

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.2.2.2 Verify the value of SR 3.2.2.2 and SR 3.2.2.3 completed SR 3.2.2.1

SR 3.2.2.1 Note

4.2.3.3 F_r shall be ~~determined~~ each time ~~a calculation of~~ F_r^T is required ~~by~~ using the incore detectors to obtain a power distribution map with all ~~full length~~ CEAs at or above the Long Term Steady State Insertion Limit ~~for the existing reactor coolant pump combination.~~

F_r^T shall be determined

as specified in the COLR

SR 3.2.2.3 Verify the value of SR 3.2.2.2 and SR 3.2.2.3 completed SR 3.2.2.1 required

SR 3.2.2.1 Note

~~4.2.3.4~~ T_q shall be ~~determined~~ each time ~~a calculation of~~ F_r^T is ~~made using a non-full core power distribution analysis code. The value of T_q used to determine F_r^T in this case shall be the measured value of T_q.~~

LA02

SR 3.2.2.2 Frequency
SR 3.2.2.3 Frequency

In accordance with the Frequency requirements of SR 3.2.2.1.

~~Pages 3/4 2-12 (Amendment 42) has been deleted from the Technical Specifications. The next page is 3/4 2-13.~~ A01

DISCUSSION OF CHANGES
ITS 3.2.2, TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 4.2.3.2.c states that CTS 4.2.3.2 be performed within 4 hours if the Azimuthal Power Tilt (Tq) is > 0.03. CTS 3.2.4 Action a. states that with the Tq > 0.03 the F_r^T be verified within limits within 4 hours if the Tq is not restored. CTS 4.2.3.2.c is redundant to CTS 3.2.4 Action a. and is deleted.

This change is designated as an administrative change and is acceptable because the change does not result in technical changes to the CTS.

A03 CTS 3.2.3 Applicability is MODE 1 with a footnote (footnote *) for MODE 1 stating "See Special Test Exception 3.10.2." ITS 3.2.2 does not contain the footnote or a reference to the Special Test Exceptions. This changes the CTS by not including footnote * in the ITS.

The purpose of the footnote references is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 4.2.3.1 states that the provisions of Specification 4.0.4 are not applicable, and thereby provides an allowance for entering the next higher MODE of Applicability when the Surveillance is not met. ITS 3.2.2 does not provide a Surveillance Note that states that the provisions of LCO 4.0.4 are not applicable. LCO 4.0.4 states that entry into a MODE or other specified condition in the Applicability of a Limiting Condition for Operation (LCO) shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by Surveillance Requirement 4.0.3. LCO 4.0.3 provides the provisions for a missed surveillance and does not apply to a surveillance known to not be met within its specified Frequency prior to entering the Mode of Applicability.

The purpose of CTS 4.2.3.1 is to provide an allowance for entering the MODE of applicability when any Surveillance is not met.

This change is designated as more restrictive because the CTS 4.0.4 MODE change allowance is deleted and entry into MODE 1 shall only be made when the LCO's surveillances have been met within their specified Frequency.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements)* CTS 3.2.3 Action b. states, in part, establish a revised upper THERMAL POWER level limit, and provides the COLR details used for this determination. ITS 3.2.2 Required Action A.3 states "Establish a revised upper THERMAL POWER limit as specified in the COLR." This changes the CTS by relocating the details for making the revised upper THERMAL POWER limit determination to the COLR.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to establish a revised upper THERMAL POWER limit as specified in the COLR. Also, this change is acceptable because these types of procedural details will be adequately controlled in the COLR. Changes to the COLR are made under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements)* CTS 4.2.3.2 and CTS 4.2.3.4 state details for calculating F_r^T with a non-full core power distribution analysis code and with a full core power distribution analysis code. ITS SR 3.2.2.1 and ITS SR 3.2.2.3 do not provide details for calculating F_r^T. This changes the CTS by relocating the details for calculating F_r^T to the ITS Bases. The ITS still retains the requirement to verify the value of F_r and to verify the value of Tq.

The removal of these details for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. The Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled.

This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.2.3 Action a. states that with F_r^T not within limits, within 6 hours be in at least HOT STANDBY within the next 6 hours as an alternative to CTS Action b. ITS 3.2.2 Action B states that with the Required Actions and associated Completion Times (of Condition A) not met, then be in MODE 2 with a Completion Time of 6 hours. This changes the CTS by only requiring that MODE 2 be entered. Once in MODE 2, LCO 3.0.2 allows the LCO to be exited since the Applicability is MODE 1, and completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications. R1

The purpose of CTS 3.2.3 Action a. and ITS 3.2.2 Action B is to reduce THERMAL POWER. Exiting the Mode of Applicability provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. R1

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.3 Action a. states that with F_r^T not within limits, within 6 hours be in at least HOT STANDBY as an alternative to CTS Action b. ITS 3.2.2 Action B states that with the Required Actions and associated Completion Times (of Condition A) not met, then be in MODE 2 with a Completion Time of 6 hours. The ITS 3.2.2 Completion Time for Required Action A.1, A.2, and A.3 is 6 hours. Therefore, up to 12 hours may be allowed to be in MODE 2. This changes the CTS by allowing 6 hours to restore F_r^T to within limits before requiring that MODE 2 be entered within 6 hours. See DOC L01 for a discussion of change from MODE 3 (CTS) to MODE 2 (ITS). R1

The purpose of CTS 3.2.3 Action a. and ITS 3.2.2 Action B is to exit the Mode of Applicability should the LCO not be restored or the Required Actions not be completed within the associated Completion Times. A Note modifying ITS 3.2.2 Condition A requires Required Actions A.1, A.2, and A.3 to be completed if the Condition is entered. This ensures that corrective action is taken prior to unrestricted operation. The limitations on F_r^T provided in the COLR ensure that the assumptions used in the analysis for establishing the ASI, LCO, and LSSS R1

remain valid during operation at the various allowable CEA group insertion limits. If F_r^T exceeds its basic limitation, operation may continue under the additional restrictions imposed by the Required Actions (reducing THERMAL POWER, withdrawing CEAs to or beyond the long term steady state insertion limits of LCO 3.1.6, and establishing a revised upper THERMAL POWER limit) because these additional restrictions provide adequate provisions to ensure the assumptions used in establishing the LHR, LCO, and LSSS remain valid. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period.

This change is designated as less restrictive because the ITS allows up to 6 hours longer to exit the Mode of Applicability should the LCO not be restored or the Required Actions not be completion within the associated Completion Times.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

5
4

3.2.~~3~~ Total Integrated Radial Peaking Factor (F_r^T) ~~(Analog)~~

2

3.2.3 LCO 3.2.~~3~~ The calculated value of F_r^T shall be within the limits specified in the COLR.

2

Applicability APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION ACTION b. A. ----------NOTE---------- Required Actions shall be completed if this Condition is entered. ---------- F_r^T not within limit.	A.1 Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within limits specified in the COLR.	6 hours
	AND	
	A.2 Withdraw the control element assemblies (CEAs) to or beyond the long term steady state insertion limits of LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits," as specified in the COLR.	6 hours
	AND	
	A.3 Establish a revised upper THERMAL POWER limit as specified in the COLR.	6 hours
ACTION a. B. Required Actions and associated Completion Times not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
SR 4.2.3.2 SR 4.2.3.2.a SR 4.2.3.2.b	SR 3.2.~~3~~.1 2	----------NOTE---------- SR 3.2.~~3~~.2 and SR 3.2.~~3~~.3 shall be completed each time SR 3.2.~~3~~.1 is required. F_r^T shall be determined by using the incore detectors to obtain a power distribution map with all ~~full length~~ CEAs at or above the long term steady state insertion limit as specified in the COLR. ---------- Verify the value of F_r^T.	Prior to operation > 70% RTP after each fuel loading AND ~~[Each 31 days of accumulated operation in MODE 1~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	4 3 2
SR 4.2.3.3	SR 3.2.~~3~~.2 2	Verify the value of F_r.	In accordance with the Frequency requirements of SR 3.2.~~3~~.1 2	4
SR 4.2.3.4	SR 3.2.~~3~~.3 2	Verify the value of T_q.	In accordance with the Frequency requirements of SR 3.2.~~3~~.1 2	4

3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

5
4

3.2.~~3~~ Total Integrated Radial Peaking Factor (F_r^T) ~~(Analog)~~

2

3.2.3

LCO 3.2.~~3~~ The calculated value of F_r^T shall be within the limits specified in the COLR.

2

Applicability

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION ACTION b. A. ----------NOTE---------- Required Actions shall be completed if this Condition is entered. ---------- F_r^T not within limit.	A.1 Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within limits specified in the COLR. AND A.2 Withdraw the control element assemblies (CEAs) to or beyond the long term steady state insertion limits of LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits," as specified in the COLR. AND A.3 Establish a revised upper THERMAL POWER limit as specified in the COLR.	6 hours 6 hours 6 hours
ACTION a. B. Required Actions and associated Completion Times not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
SR 4.2.3.2 SR 4.2.3.2.a SR 4.2.3.2.b	SR 3.2.~~3~~.1 2	----------NOTE---------- SR 3.2.~~3~~.2 and SR 3.2.~~3~~.3 shall be completed each time SR 3.2.~~3~~.1 is required. F_r^T shall be determined by using the incore detectors to obtain a power distribution map with all ~~full length~~ CEAs at or above the long term steady state insertion limit as specified in the COLR. ---------- Verify the value of F_r^T.	Prior to operation > 70% RTP after each fuel loading AND ~~[Each 31 days of accumulated operation in MODE 1~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	4 3 2
SR 4.2.3.3	SR 3.2.~~3~~.2 2	Verify the value of F_r.	In accordance with the Frequency requirements of SR 3.2.~~3~~.1 2	4
SR 4.2.3.4	SR 3.2.~~3~~.3 2	Verify the value of T_q.	In accordance with the Frequency requirements of SR 3.2.~~3~~.1 2	4

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.2, TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ISTS SR 3.2.2.1 Note, deletes the term "full length" as related to control element assemblies (CEAs). Part length CEAs are not used at PSL Unit 1 and PSL Unit 2.

4. The ISTS includes ISTS 3.2.1, Linear Heat Rate (LHR), ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^T), ISTS 3.2.3, Total Integrated Radial Peaking Factor (F_r^T), ISTS 3.2.4, Azimuthal Power Tilt (T_q), and ISTS 3.2.5, Axial Shape Index (ASI). CTS do not include a Specification for ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^T). The CTS and ISTS are renumbered. CTS 3.2.3, Total Integrated Radial Peaking Factor (F_r^T), CTS 3.2.4, Azimuthal Power Tilt (T_q), and CTS 3.2.5, Axial Shape Index (ASI), are renumbered as ITS 3.2.2, ITS 3.2.3, and ITS 3.2.4, respectively.

5. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

(5) (4)

B 3.2.~~3~~ (2) Total Integrated Radial Peaking Factor (F_r^T) ~~(F_{XY}^T) (Analog)~~

BASES

BACKGROUND

The purpose of this LCO (Total Integrated Radial Peaking Factor (F^T_r)) is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. The use of CEAs to alter the axial power distribution,

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution, and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings (LSSS) and this LCO are based on the accident analyses (Ref~~s~~. 1 ~~and 2~~), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the linear heat rate (LHR) and departure from nucleate boiling (DNB).

BACKGROUND (continued)

LINEAR HEAT RATE ( ) AZIMUTHAL POWER TILT ( )
AXIAL SHAPE INDEX ( )

The limits on LHR, ~~Total Planar Radial Peaking Factor (F_{XY}^T),~~ F_r^T, T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis. 4 1

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and ~~are~~ is capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring ~~the~~ ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR. 1

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following conditions are assumed:

a. The CEA insertion limits of LCO 3.1.5, "Shutdown CEA Insertion Limits," and LCO 3.1.6, "Regulating CEA Insertion Limits," are satisfied,

b. The T_q restrictions of LCO 3.2.~~4~~ 3 are satisfied, and

c. ~~F_{xy}^T~~ F_r^T does not exceed the limits of LCO 3.2.2. 4

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms established for the individual incore detector segments ensure that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances~~, set in conservative directions, for:~~ described in Reference 3. 3

~~a. A measurement calculational uncertainty factor of 1.062,~~

~~b. An engineering uncertainty factor of 1.03,~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion, and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02~~.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and AOOs (Condition 2) (Ref. ~~3~~, GDC 10). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria: 4 1

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. ~~4~~), 1 5

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. ~~3~~, GDC 10), 1 4

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 2~~8~~0 cal/gm (Ref. ~~[~~ ~~]~~), and 2 3 2

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. ~~3~~, GDC 26). 1 4

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. ~~4~~). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution. 1 5

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. ~~4~~). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction. 1 5

The LCOs governing LHR, ASI, and the Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI, ~~F_{XY}^T~~, and F_r^T limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analysis. 4

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not normally occur while the reactor is operating at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution cause increased power peaking and correspondingly increased local LHR. 1

F^T_r satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO limits for certain power distribution are based on correlations between power peaking and measured variables used as inputs to LHR and DNB ratio operating limits. The LCO limits for power distribution, except T_q, are provided in the COLR. The limitation on the LHR ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F. 1

APPLICABILITY

In MODE 1, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because ~~there~~ is not sufficient THERMAL POWER to require a limit on the core power distribution. 1

ACTIONS

A.1, A.2, and A.3

A Note modifying Condition A requires Required Actions A.1, A.2, and A.3 to be completed if the Condition is entered. This ensures that corrective action is taken prior to unrestricted operation.

The limitations on F_r^T provided in the COLR ensure that the assumptions used in the analysis for establishing the ASI, LCO, and LSSS remain valid during operation at the various allowable CEA group insertion limits. If F_r^T exceeds its basic limitation, operation may continue under the additional restrictions imposed by the Required Actions (reducing THERMAL POWER, withdrawing CEAs to or beyond the long term steady state insertion limits of LCO 3.1.6, and establishing a revised upper THERMAL POWER limit) because these additional restrictions provide adequate provisions to ensure that the assumptions used in establishing the LHR, LCO, and LSSS remain valid. Six hours to return F_r^T to within its limits by adjusting the ASI limits based on maximum power allowed for F^T_r ~~F^T_{XY}~~ is reasonable and ensures that all CEAs meet the long term steady state insertion limits of LCO 3.1.6. 4

ACTIONS (continued)

B.1

If F_r^T cannot be returned to within its limit, THERMAL POWER must be reduced. A change to MODE 2 provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.~~3~~.1 (2) 5

The periodic Surveillance to determine the calculated F_r^T ensures that F_r^T remains within the range assumed in the analysis throughout the fuel cycle. Determining the measured F_r^T once after each fuel loading prior to exceeding 70% RTP ensures that the core is properly loaded.

~~[Performance of the Surveillance every 31 days of accumulated operation in MODE 1 provides reasonable assurance that unacceptable changes in the F_r^T are promptly detected.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

The power distribution map can only be obtained after THERMAL POWER exceeds 20% RTP because the incore detectors are not reliable below 20% RTP.

The SR is modified by a Note that requires SR 3.2.3.2 and SR 3.2.3.3 be completed each time SR 3.2.3.1 is completed. This procedure is required because the values computed by these SRs are required to perform this SR.

F_r^T is calculated by the expression $F_r^T = Fr\,(1 + Tq)$ when Fr is calculated with a non-full core power distribution analysis code and is calculated by the expression $F_r^T = Fr$ when calculations are performed with a full core power distribution analysis code. 6

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.~~3~~.2 and SR 3.2.~~3~~.3

Measuring the values of $F_r^{~~X~~}$ and T_q each time a value of F_r^T is calculated ensures that the calculated value of F_r^T accurately reflects the condition of the core.

The Frequency for these Surveillances is in accordance with the requirements of SR 3.2.~~3~~.1 because these SRs provide information to complete SR 3.2.2.1.

REFERENCES

1. FSAR, Chapter ~~[~~15~~]~~.
2. ~~FSAR, Chapter [6]~~. NUREG 0800, Section 4.2, Appendix B
3. Technical Requirements Manual
~~3~~. 10 CFR 50, Appendix A.
~~4~~. 10 CFR 50.46.

B 3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~ (5) (4)

B 3.2.~~3~~ (2) Total Integrated Radial Peaking Factor (F_{XY}^T) ~~(Analog)~~ (F_r^T)

BASES

BACKGROUND

The purpose of this LCO (Total Integrated Radial Peaking Factor (F^T_r)) is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. The use of CEAs to alter the axial power distribution,

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution, and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings (LSSS) and this LCO are based on the accident analyses (Ref~~s~~. 1 ~~and 2~~), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the linear heat rate (LHR) and departure from nucleate boiling (DNB).

BACKGROUND (continued)

LINEAR HEAT RATE ( ) AZIMUTHAL POWER TILT ( )
AXIAL SHAPE INDEX ( )

The limits on LHR, ~~Total Planar Radial Peaking Factor (F_{XY}^T),~~ F_r^T, T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and ~~are~~ is capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring ~~the~~ ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following conditions are assumed:

a. The CEA insertion limits of LCO 3.1.5, "Shutdown CEA Insertion Limits," and LCO 3.1.6, "Regulating CEA Insertion Limits," are satisfied,

b. The T_q restrictions of LCO 3.2.~~4~~ 3 are satisfied, and

c. ~~F_{xy}^T~~ F_r^T does not exceed the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms established for the individual incore detector segments ensure that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances~~, set in conservative directions, for:~~ described in Reference 3.

~~a. A measurement calculational uncertainty factor of 1.062,~~

~~b. An engineering uncertainty factor of 1.03,~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion, and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02~~.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and AOOs (Condition 2) (Ref. ~~3~~, GDC 10). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria: 4 1

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. ~~4~~), 1
5

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. ~~3~~, GDC 10), 1
4

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 2~~8~~0 cal/gm (Ref. ~~[~~ ~~]~~), and 2
3 2

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. ~~3~~, GDC 26). 1
4

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. ~~4~~). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution. 1
5

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. ~~4~~). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction. 1
5

The LCOs governing LHR, ASI, and the Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI, ~~F_{XY}^T~~, and F_r^T limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analysis. 4

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not normally occur while the reactor is operating at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution cause increased power peaking and correspondingly increased local LHR.

F^T_r satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO limits for certain power distribution are based on correlations between power peaking and measured variables used as inputs to LHR and DNB ratio operating limits. The LCO limits for power distribution, except T_q, are provided in the COLR. The limitation on the LHR ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because ~~there~~ is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1, A.2, and A.3

A Note modifying Condition A requires Required Actions A.1, A.2, and A.3 to be completed if the Condition is entered. This ensures that corrective action is taken prior to unrestricted operation.

The limitations on F_r^T provided in the COLR ensure that the assumptions used in the analysis for establishing the ASI, LCO, and LSSS remain valid during operation at the various allowable CEA group insertion limits. If F_r^T exceeds its basic limitation, operation may continue under the additional restrictions imposed by the Required Actions (reducing THERMAL POWER, withdrawing CEAs to or beyond the long term steady state insertion limits of LCO 3.1.6, and establishing a revised upper THERMAL POWER limit) because these additional restrictions provide adequate provisions to ensure that the assumptions used in establishing the LHR, LCO, and LSSS remain valid. Six hours to return F_r^T to within its limits by adjusting the ASI limits based on maximum power allowed for F_r^T ~~F^T_{XY}~~ is reasonable and ensures that all CEAs meet the long term steady state insertion limits of LCO 3.1.6.

BASES

ACTIONS (continued)

B.1

If F_r^T cannot be returned to within its limit, THERMAL POWER must be reduced. A change to MODE 2 provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.~~3~~.1
2

5

The periodic Surveillance to determine the calculated F_r^T ensures that F_r^T remains within the range assumed in the analysis throughout the fuel cycle. Determining the measured F_r^T once after each fuel loading prior to exceeding 70% RTP ensures that the core is properly loaded.

~~[Performance of the Surveillance every 31 days of accumulated operation in MODE 1 provides reasonable assurance that unacceptable changes in the F_r^T are promptly detected.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

The power distribution map can only be obtained after THERMAL POWER exceeds 20% RTP because the incore detectors are not reliable below 20% RTP.

The SR is modified by a Note that requires SR 3.2.3.2 and SR 3.2.3.3 be completed each time SR 3.2.3.1 is completed. This procedure is required because the values computed by these SRs are required to perform this SR.

F_r^T is calculated by the expression $F_r^T = Fr\ (1 + Tq)$ when Fr is calculated with a non-full core power distribution analysis code and is calculated by the expression $F_r^T = Fr$ when calculations are performed with a full core power distribution analysis code.

6

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.~~3~~.2 and SR 3.2.~~3~~.3

Measuring the values of $F_r^{\prime}$ and T_q each time a value of F_r^T is calculated ensures that the calculated value of F_r^T accurately reflects the condition of the core.

The Frequency for these Surveillances is in accordance with the requirements of SR 3.2.~~3~~.1 because these SRs provide information to complete SR 3.2.2.1.

REFERENCES

1. FSAR, Chapter ~~[~~15~~]~~.
2. ~~FSAR, Chapter [6].~~ NUREG 0800, Section 4.2, Appendix B
3. Technical Requirements Manual
~~3~~. 10 CFR 50, Appendix A.
~~4~~. 10 CFR 50.46.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.2, BASES, TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS contains tolerances to be applied to the incore detector alarm setpoints. PSL Unit 1 UFSAR, Section 13.8, Licensee-Controlled Technical Specification Requirements, and PSL Unit 2 UFSAR, Section 13.7, Licensee-Controlled Technical Specification Requirements, each contain these tolerances to be applied to the incore detector alarm setpoints. Additionally, these tolerances will be provided in the Technical Requirements Manual. Therefore, the tolerances to be applied to the incore detector alarm setpoints are deleted.

4. The ISTS includes ISTS 3.2.1, Linear Heat Rate (LHR), ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^T), ISTS 3.2.3, Total Integrated Radial Peaking Factor (F_r^T), ISTS 3.2.4, Azimuthal Power Tilt (T_q), and ISTS 3.2.5, Axial Shape Index (ASI). CTS do not include a Specification for ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^T). The CTS and ISTS are renumbered. CTS 3.2.3, Total Integrated Radial Peaking Factor (F_r^T), CTS 3.2.4, Azimuthal Power Tilt (T_q), and CTS 3.2.5, Axial Shape Index (ASI), are renumbered as ITS 3.2.2, ITS 3.2.3, and ITS 3.2.4, respectively.

5. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

6. CTS 4.2.3.2 and CTS 4.2.3.4 state details for calculating F_r^T with a non-full core power distribution analysis code and with a full core power distribution analysis code. ITS SR 3.2.2.1 and ITS SR 3.2.2.3 do not provide details for calculating F_r^T. This changes the CTS by relocating the details for calculating F_r^T to the ITS Bases. This change is acceptable since procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications and relocating these details to the ITS Bases.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.2, TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

ITS 3.2.3, AZIMUTHAL POWER TILT (T_q)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

POWER DISTRIBUTION LIMITS

AZIMUTHAL POWER TILT – T_q

LIMITING CONDITION FOR OPERATION

3

LCO 3.2.3 3.2.~~4~~ ~~The AZIMUTHAL POWER TILT (~~T_q~~) shall not exceed~~ 0.03.

be ≤

Applicability **APPLICABILITY:** MODE 1*.

with THERMAL POWER > 50% RTP.

A02

L01

ACTION:

ACTION A
Required Action A.1

a. ~~With the~~ indicated ~~AZIMUTHAL POWER TILT determined to be~~ > .030 ~~but~~ ≤ 0.10, either ~~correct the power tilt~~ within two hours or determine within ~~the next~~ 2 hours and ~~at least~~ once per ~~subsequent~~ 8 hours, ~~that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (~~F_r^T~~) is within the limits of Specification 3.2.3.~~

and

Tq

Restore Tq to ≤ 0.03

thereafter

Required Action A.2

Verify

LCO 3.2.2, "Total Integrated Radial Peaking Factor (F^T_r)."

M02

ACTION B
Required Action B.1

b. ~~With the indicated AZIMUTHAL POWER TILT determined~~ to ~~be >~~ 0.10, ~~operation may proceed for up to~~ 2 hours ~~provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (~~F_r^T~~) is within the limits of Specification~~ 3.2.~~3~~. ~~Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination~~.

Restore Tq

≤

within

Required Action B.2

2 within 1 hour

Verify

LCO

ACTION C
Required Action C.1

Required Action and associated Completion Time not met. Reduce THERMAL POWER to ≤ 50% RTP within 4 hours.

L02

L01

SURVEILLANCE REQUIREMENT

~~4.2.4.1 The provisions of Specification 4.0.4 are not applicable.~~

M01

Verify Tq is

s

SR 3.2.3.1 ~~4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be~~ within ~~the~~ limit ~~by:~~

~~a. Calculating the tilt~~ in accordance with the Surveillance Frequency Control Program ~~when the Subchannel Deviation Alarm is OPERABLE,~~

~~* See Special Test Exception 3.10.2.~~

A02

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.2.3.1 Frequency

~~b. Calculating the tilt at least~~ once per 12 hours when the Subchannel Deviation Alarm is inoperable~~,~~ and

SR 3.2.3.1 Frequency

~~c. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least~~ once per 12 hours when one excore channel is inoperable and ~~THERMAL POWER is~~ > 75% ~~of RATED THERMAL POWER.~~

RTP

A01

POWER DISTRIBUTION LIMITS

3/4.2.4 AZIMUTHAL POWER TILT – T_q

LIMITING CONDITION FOR OPERATION

LCO 3.2.3

3.2.~~4~~ ~~The AZIMUTHAL POWER TILT (T_q)~~ shall ~~not exceed~~ 0.03.

Applicability

APPLICABILITY: MODE 1*.

ACTION:

ACTION A
Required Action A.1

~~a.~~ ~~With the~~ indicated ~~AZIMUTHAL POWER TILT determined to be~~ > .030 ~~but~~ ≤ 0.10, either ~~correct the power tilt~~ within 2 hours or determine within ~~the next~~ 2 hours and ~~at least~~ once per ~~subsequent~~ 8 hours, ~~that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T)~~ is within the limits of ~~Specification 3.2.3.~~

Required Action A.2

ACTION B
Required Action B.1

~~b.~~ ~~With the indicated AZIMUTHAL POWER TILT determined to be > 0.10,~~ ~~operation may proceed for up to~~ 2 hours ~~provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T)~~ is within the limits of ~~Specification~~ 3.2.~~3~~. ~~Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.~~

Required Action B.2

ACTION C
Required Action C.1

Required Action and associated Completion Time not met. Reduce THERMAL POWER to ≤ 50% RTP within 4 hours.

SURVEILLANCE REQUIREMENTS

~~4.2.4.1~~ The provisions of Specification 4.0.4 are not applicable.

SR 3.2.3.1

~~4.2.4.2~~ ~~The AZIMUTHAL POWER TILT shall be determined to be~~ within ~~the~~ limit ~~by:~~

~~a. Calculating the tilt~~ in accordance with the Surveillance Frequency Control Program.

~~b. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least~~ once per 12 hours when one excore channel is inoperable and ~~THERMAL POWER is~~ > 75% ~~of RATED THERMAL POWER~~.

~~* See Special Test Exception 3.10.2.~~

DISCUSSION OF CHANGES
ITS 3.2.3, AZIMUTHAL POWER TILT (T_q)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.2.4 Applicability is MODE 1 with a footnote (footnote *) for MODE 1 stating "See Special Test Exception 3.10.2." ITS 3.2.3 does not contain the footnote or a reference to the Special Test Exceptions. This changes the CTS by not including footnote * in the ITS.

The purpose of the footnote references is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 4.2.4.1 states that the provisions of Specification 4.0.4 are not applicable, and thereby provides an allowance for entering the next higher MODE of Applicability when the Surveillance is not met. ITS 3.2.3 does not provide a Surveillance Note that states that the provisions of LCO 4.0.4 are not applicable. LCO 4.0.4 states that entry into a MODE or other specified condition in the Applicability of a Limiting Condition for Operation (LCO) shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by Surveillance Requirement 4.0.3. LCO 4.0.3 provides the provisions for a missed surveillance and does not apply to a surveillance known to not be met within its specified Frequency prior to entering the Mode of Applicability.

The purpose of CTS 4.2.4.1 is to provide an allowance for entering the MODE of applicability when any Surveillance is not met. The ITS MODE change is changed from MODE 1 to MODE 1 with THERMAL POWER > 50% RTP. See DOC L01 for a discussion of change regarding the MODE change from MODE 1, to MODE 1 with THERMAL POWER > 50% RTP. This change allows the performance of ITS 3.2.3.1 after entry into MODE 1 and up to 50% RTP. The ITS MODE change to MODE 1 with THERMAL POWER > 50% is less restrictive; however, the deletion of CTS 4.0.4 is solely a more restrictive change.

This change is designated as more restrictive because the CTS 4.0.4 MODE change allowance is deleted and entry into MODE 1 shall only be made when the LCO's surveillances have been met within their specified Frequency.

DISCUSSION OF CHANGES
ITS 3.2.3, AZIMUTHAL POWER TILT (T_q)

M02 CTS 3.2.4.a states that T_q be corrected within 2 hours or F_r^T be verified within limits within the next 2 hours and at least once per subsequent 8 hours. ITS 3.2.3 Condition A states that T_q be restored to ≤ 0.03 within 2 hours or F_r^T be verified within the limits of LCO 3.2.2 within 2 hours and once per 8 hours thereafter. CTS requires the first verification that F_r^T is within limits be completed within 4 hours. ITS requires the first verification that F_r^T is within limits be completed within 2 hours.

The purpose of CTS/ITS is to provide reasonable assurance that the core is operating within its thermal limits during power operation beyond 2 hours by verifying that F_r^T is within limits if T_q is not corrected. The change requires that for operation to continue beyond 2 hours with T_q > 0.03 and ≤ 0.10, F_r^T must be verified within limits within 2 hours. Two hours is sufficient time to allow the operator to reposition CEAs, and significant radial xenon redistribution cannot occur within this time. This change is consistent with the ISTS Required Action A.2 Completion Time.

This change is designated as more restrictive as it requires that with T_q > 0.03 and ≤ 0.10, the first verification that F_r^T is within limits be completed within 2 hours.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.2.4 states that the Applicability is MODE 1. ITS 3.2.3 states that the Applicability is MODE 1 with THERMAL POWER > 50% RTP. This changes the CTS by relaxing the Applicability to MODE 1 with THERMAL POWER > 50% RTP.

The purpose of CTS 3.2.4 Applicability is to state the MODE in which Tq must be maintained within the limits assumed in accident analysis to ensure that fuel damage does not result following an AOO. ITS 3.2.3 relaxes the Applicability to MODE 1 with THERMAL POWER > 50% RTP. CTS 4.2.4.1 states that the provisions of Specification 4.0.4 are not applicable, and thereby provides an allowance for entering the next higher MODE of Applicability with surveillances not met. ITS 3.2.3 Applicability of MODE 1 with THERMAL POWER > 50% RTP allows entry into MODE 1 with surveillances not met. See DOC M01 for the discussion of change for the MODE change.

This change limits the applicability of T_q to power operations with THERMAL POWER > 50% RTP. This change clarifies the applicability of the T_q, Azimuthal Power Tilt, consistent with the industry Standard Technical Specification (STS), approved by the NRC for generic use as NUREG-1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS). This change is acceptable because the LCO requirements applied in ITS MODE 1 continue to ensure process variables, structures, systems, and components are maintained in the conditions assumed in the safety analyses and licensing basis, and the proposed TS has been generically approved for industry usage.

In MODE 1 with THERMAL POWER > 50% RTP, T_q must be maintained within the limits assumed in accident analysis to ensure that fuel damage does not result following an anticipated operational occurrence (AOO). In other MODES, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on T_q, and xenon transients generated within the lower power level range are not severe. In addition, significant margin to thermal limits exists at lower power levels and therefore thermal limits are not significantly challenged.

The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the fuel design criteria. The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of the power distribution LCOs, including ITS 3.2.3, T_q. This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analysis.

PSL controls the AXIAL SHAPE INDEX (ASI) during startup, steady-state, and transients such as a dropped CEA. ASI COLR bands are applicable above 40% RTP while fuel conditioning ASI bands are applicable above 50% RTP. Axial shape control establishes and maintains axial shape within the fuel conditioning control bands prior to exceeding 50% RTP. The intent of the fuel preconditioning guidelines is to minimize fuel pellet / clad interaction and maximize fuel rod integrity. When THERMAL POWER is ≥ 50 RTP, boration or dilution is the primary means to compensate for changes in power level and transient xenon. Using boration/dilution to compensate for power changes subjects the fuel rods to uniform and smooth power transients. Regulating Group CEAs are used to respond deviations of ASI from its target equilibrium shape index.

This change is designated as less restrictive because the LCO requirements will not be applied under plant conditions in which they would be applied under the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.2.4 Action b. states, in part, “with in the indicated Azimuthal Power Tilt determined to be > 0.10 operation may proceed for up to 2 hours”. CTS 3.2.4 CTS 3.2.4.b states, in part, the subsequent operation be “≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.” PSL only operates in the four reactor coolant pump combination; therefore, subsequent operation is allowable provided the THERMAL POWER is reduced to ≤ 20% RTP – the

equivalent THERMAL POWER for the CTS. ITS Required Action B.2 states "Restore T_q to ≤ 0.10 within 2 hours. If T_q is not restored to ≤ 0.10 within 2 hours, Required Action C.1 requires reducing THERMAL POWER to the ≤ 50% RTP within 4 hours. See DOC L01 for a discussion of change regarding the ITS MODE change from MODE 1, to MODE 1 with THERMAL POWER > 50% RTP.

The purpose of CTS 3.2.4 Action b. is to reduce THERMAL POWER to ≤ 20% if T_q is not restored to ≤ 0.10 within 2 hours. ITS requires that with T_q not restored to ≤ 0.10 within 2 hours, reduce THERMAL POWER to the ≤ 50% RTP within 4 hours.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

3.2.4 3.2.~~4~~ AZIMUTHAL POWER TILT (T_q) ~~(Analog)~~ 3 7

LCO 3.2.4 LCO 3.2.~~4~~ T_q shall be ≤ ~~[~~0.03~~]~~. 3 2

Applicability APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a. A. Indicated T_q > ~~[~~0.03~~]~~ and ≤ 0.10.	A.1 Restore T_q to ≤ ~~[~~0.03~~]~~.	2 hours 2
	OR	
	A.2 Verify ~~F_{xy}^T and~~ F_r^T ~~are~~ (is) within the limits of LCO 3.2.2, ~~"Total Planar Radial Peaking Factor (F_{xy}^T)," and LCO 3.2.3,~~ "Total Integrated Radial Peaking Factor (F_r^T)~~;~~."	2 hours AND Once per 8 hours thereafter 3
~~B~~. (C) Required Action and associated Completion Time ~~of Condition A~~ not met.	~~B~~.1 (C) Reduce THERMAL POWER to ≤ 50% RTP.	4 hours 4 5

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b. ~~C~~. Indicated T_q > 0.10. (B)	~~NOTE~~ ~~All subsequent Required Actions must be completed if power reduction commences prior to restoring $T_q \leq 0.10$.~~	
	~~C~~.1 (B) Verify F_{xy}^T ~~and~~ F_r^T ~~are~~ (is) within the limits of LCO 3.2.2 ~~and LCO 3.2.3, respectively~~.	1 hour
	AND	
	~~C.2~~ ~~Reduce THERMAL POWER to < 50% RTP.~~	~~2 hours~~
	~~AND~~	
	~~C.3~~ (B.2) Restore T_q to ≤ ~~[0.03]~~ (0.10).	2 hours ~~Correct the cause of the out of limit condition prior to increasing THERMAL POWER. Subsequent power operation above 50% RTP may proceed provided that the measured T_q is verified ≤ [0.03] at least once per hour for 12 hours, or until verified at 95% RTP~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 4.2.4.2.a	SR 3.2.~~4~~.1 (3)	Verify T_q is within limits.	12 hours when the subchannel deviation alarm is inoperable (3) (2) AND 12 hours when one excore detector is inoperable and > 75% RTP AND ~~[12 hours~~ ~~OR~~ (2) In accordance with the Surveillance Frequency Control Program ~~]~~

3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

3.2.4 3.2.~~4~~ AZIMUTHAL POWER TILT (T_q) ~~(Analog)~~

3 7

LCO 3.2.4 LCO 3.2.~~4~~ T_q shall be ≤ ~~[~~0.03~~]~~.

3 2

Applicability APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a. A. Indicated T_q > ~~[~~0.03~~]~~ and ≤ 0.10.	A.1 Restore T_q to ≤ ~~[~~0.03~~]~~.	2 hours
	OR	
	A.2 Verify ~~F_{xy}^T and~~ F_r^T ~~are~~ (is) within the limits of LCO 3.2.2, ~~"Total Planar Radial Peaking Factor (F_{xy}^T)," and LCO 3.2.3,~~ "Total Integrated Radial Peaking Factor (F_r^T)~~;~~."	2 hours AND Once per 8 hours thereafter
~~B~~. (C) Required Action and associated Completion Time ~~of Condition A~~ not met.	~~B~~.1 (C) Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

2 3 4 5

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b. ~~C~~. Indicated T_q > 0.10.	~~NOTE~~ ~~All subsequent Required Actions must be completed if power reduction commences prior to restoring $T_q \leq 0.10$.~~	
	~~C~~.1 Verify F_{xy}^T ~~and~~ F_r^T ~~are~~ is within the limits of LCO 3.2.2 ~~and LCO 3.2.3, respectively~~.	1 hour
	AND	
	~~C.2~~ ~~Reduce THERMAL POWER to < 50% RTP.~~	~~2 hours~~
	~~AND~~	
	~~C.3~~ Restore T_q to ≤ ~~[0.03]~~ 0.10.	2 hours ~~Correct the cause of the out of limit condition prior to increasing THERMAL POWER. Subsequent power operation above 50% RTP may proceed provided that the measured T_q is verified ≤ [0.03] at least once per hour for 12 hours, or until verified at 95% RTP~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 4.2.4.2.a	SR 3.2.~~4~~.1 3	Verify T_q is within limits.	12 hours when one excore detector is inoperable and > 75% RTP 3 AND 2 ~~[12 hours~~ ~~OR~~ 2 In accordance with the Surveillance Frequency Control Program ~~]~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.3, AZIMUTHAL POWER TILT (T_q)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS includes ISTS 3.2.1, Linear Heat Rate (LHR), ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^T), ISTS 3.2.3, Total Integrated Radial Peaking Factor (F_r^T), ISTS 3.2.4, Azimuthal Power Tilt (T_q), and ISTS 3.2.5, Axial Shape Index (ASI). CTS do not include a Specification for ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^T). The CTS and ISTS are renumbered. CTS 3.2.3, Total Integrated Radial Peaking Factor (F_r^T), CTS 3.2.4, Azimuthal Power Tilt (T_q), and CTS 3.2.5, Axial Shape Index (ASI), are renumbered as ITS 3.2.2, ITS 3.2.3, and ITS 3.2.4, respectively.

4. ISTS Condition B is renumbered as ITS Condition C, and ISTS Condition C is renumbered as ITS Condition B. ITS Condition C deletes the verbiage "of Condition A" so ITS Condition C and Required Action C.1 apply to both ITS Condition A and ITS Condition B.

5. CTS 3.2.4.b allows operation for up to 2 hours with $T_q > 0.10$ provided F_r^T is verified within limits. ISTS Required Action C.3 (renumbered as ITS Required Action B.2) is changed to restore T_q to ≤ 0.10 with a Completion Time of 2 hours. ISTS Required Action C.2 is deleted since this action is moved to ITS Condition C and ITS Required Action C.1. This change is like Calvert Cliffs – Unit 1, Calvert Cliffs – Unit 2 Improved Technical Specifications (ADAMS Accession No. ML052720276). JFD-4 changes ISTS Condition B to Condition C.

   If the Required Actions and associated Completion Times of ITS Condition A or Condition B are not met, Condition C must be entered. ITS Required Action C.1 requires a reduction in THERMAL POWER to ≤ 50% RTP within 4 hours. If the T_q cannot be restored to within its specified limit, THERMAL POWER must be reduced. The reduction in THERMAL POWER to ≤ 50% RTP provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition.

6. ISTS Condition C Required Actions are modified by a Note that requires all subsequent actions to be performed once power reduction commences after entering the Condition if Tq is not restored to < 0.10. This Note is deleted.

   ISTS Required Action C.3 is modified with a Note to indicate that the cause of the out of limit condition must be corrected prior to increasing THERMAL POWER, and that subsequent power operation may proceed provided that the measured Tq is verified within its limit at least once per hour for 12 hours, or until verified at 95% RTP. This Note is deleted.

If ITS Required Action B.1 or Required Action B.2 is not met within its associated Completion Time, ITS Required Action C.1 requires a reduction in THERMAL POWER to ≤ 50% RTP within 4 hours. If F_r^T is verified within limits within 1 hour, and T_q is restored to ≤ 0.10 with 2 hours, Condition A is entered, and the Required Actions are taken. If the Condition A Required Actions are not met, ITS Required Action C.1 requires a reduction in THERMAL POWER to ≤ 50% RTP within 4 hours. The reduction in THERMAL POWER to ≤ 50% RTP provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition.

7. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

B 3.2.~~4~~ AZIMUTHAL POWER TILT (T_q) ~~(Analog)~~

BASES

BACKGROUND

The purpose of this LCO (AZIMUTHAL POWER TILT (T_q)) is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution,

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution, and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Ref~~s~~. 1 ~~and 2~~), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits for linear heat rate (LHR) and departure from nucleate boiling (DNB).

BACKGROUND (continued)

The limits on LHR, ~~Total Planar Radial Peaking Factor (F_{XY}^T),~~ Total Integrated Radial Peaking Factor (F_r^T), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LCO limits are not exceeded. The Excore Detector Monitoring System performs this function by continuously monitoring ASI with OPERABLE quadrant symmetric excore neutron detectors and by verifying ASI is maintained within the limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following assumptions are made:

a. The CEA insertion limits of LCO 3.1.5, "Shutdown CEA Insertion Limits," and LCO 3.1.6, "Regulating CEA Insertion Limits," are satisfied,

b. The T_q restrictions of LCO 3.2.~~4~~ are satisfied, and

c. ~~F_{xy}^T~~ F_r^T does not exceed the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms ~~that have been~~ established for the individual incore detector segments ensure that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances~~, set in conservative directions, for:~~ described in Reference 3.

~~a. A measurement calculational uncertainty factor of 1.062,~~

~~b. An engineering uncertainty factor of 1.03,~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion, and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02~~.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or AOOs (Condition 2) (Ref. ~~3~~, GDC 10). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. ~~4~~),

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. ~~3~~, GDC 10),

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 2~~8~~0 cal/gm (Ref. ~~[ ]~~), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. ~~3~~, GDC 26).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. ~~4~~). This process is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analysis (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate ~~(LHGR)~~ so that the peak cladding temperature does not exceed 2200°F (Ref. ~~4~~). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI~~,~~ ~~F_{XY}^T~~, and F_r^T limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analyses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This potential for fuel cladding damage exists because

Fuel cladding damage does not normally occur while the reactor is operating at conditions outside these LCOs during ~~otherwise~~ normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. Changes in the power distribution cause increased power peaking and correspondingly increased local LHRs.

The T_q satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

certain

The power distribution LCO limits are based on correlations between power peaking and ~~the~~ measured variables used as inputs to the LHR and DNB operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limits on LHR ensure that in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1 with THERMAL POWER > 50% RTP, T_q must be maintained within the limits assumed in accident analysis to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on T_q.

ACTIONS

A.1 and A.2

≤

within 2 hours

If the measured T_q is > ~~[~~0.03~~]~~ and ~~<~~ 0.10, the calculation of T_q may be nonconservative. T_q must be restored within 2 hours or F^T_{XY} ~~and~~ F^T_r must be determined to be within the limits of LCO 3.2.2 ~~and LCO 3.2.3~~, and determined to be within these limits every 8 hours thereafter, as long as T_q is out of limits. Two hours is sufficient time to allow the operator to reposition CEAs, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in F^T_{XY} ~~and~~ F^T_r can be identified before the limits of LCO 3.2.2 ~~and LCO 3.2.3, respectively,~~ are exceeded.

C

~~B~~.1

conservative protection from increased peaking due to potential xenon redistribution and

or B

If Required Actions and associated Completion Times of Condition A are not met, THERMAL POWER must be reduced to ≤ 50% RTP. This requirement provides reasonable assurance that the core is operating within its thermal limits ~~and places the core in a conservative condition~~. Four hours is a reasonable time to reach 50% RTP in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

~~C~~.1, ~~C.2,~~ and ~~C.3~~

With T_q > 0.10, ~~F^T_{XY} and~~ F^T_r must be within ~~their~~ specified limits to ensure that acceptable flux peaking factors are maintained. Based on operating experience, 1 hour is sufficient time for the operator to evaluate these factors. If ~~F^T_{XY} and~~ F^T_r ~~are~~ within limits, operation may proceed for a total of 2 hours after the Condition is entered while attempts are made to restore T_q to within its limit.

If T_q ≤ 0.10 cannot be achieved, power must be reduced to ≤ 50% RTP within ~~2~~ hours. ~~If the tilt is generated due to a CEA misalignment, operating at ≤ 50% RTP allows for the recovery of the CEA. Except as a result of CEA misalignment, T_q > 0.10 is not expected; if it occurs, continued operation of the reactor may be necessary to discover the cause of the tilt. If this procedure is followed, operation is restricted to only those conditions required to identify the cause of the tilt. It is necessary to account explicitly for power asymmetries because the radial power peaking factors used in core power distribution calculations are based on an untilted power distribution.~~

~~If T_q is not restored to within its limits,~~ the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation that causes increased LHRs when the xenon redistributes. ~~If T_q cannot be restored to within its limits within 2 hours, reactor power must be reduced.~~ Reducing THERMAL POWER to ≤ 50% RTP within ~~2~~ hours provides conservative protection from increased peaking due to potential xenon redistribution. ~~The Required Actions are modified by a Note that requires all subsequent actions to be performed once power reduction commences after entering the Condition if T_q is not restored to < 0.10. This procedure ensures corrective action is taken before unrestricted power operation resumes. Following THERMAL POWER reduction to ≤ 50% RTP, T_q must be restored to ≤ [0.03] before THERMAL POWER is increased (Required Action C.3). This Required Action prevents the operator from increasing THERMAL POWER above the conservative limit when the Condition, T_q outside its limits, has existed but allows the unit to continue operation for diagnostic purposes. The Completion Time of Required Action C.3 is modified with a Note to indicate that the cause of the out of limit condition must be corrected prior to increasing THERMAL POWER. This Note also indicates that subsequent power operation above 50% RTP may proceed provided that the measured T_q is verified ≤ [0.03] at least once per hour for 12 hours, or until verified at 95% RTP. This ensures that the power distribution is responding as predicted. The Completion Time of 12 hours is a historical value that allows an acceptable exit from the LCO after the T_q value is verified acceptable for 12 hours or until 95% RTP is reached.~~

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.~~4~~.1

~~[T_q must be calculated at 12 hour intervals. The 12 hour Frequency prevents significant xenon redistribution between Surveillances.~~

~~OR~~

Tq must be determined at 12 hour intervals when the subchannel deviation alarm is inoperable, and at 12 hour intervals when one excore detector is inoperable and > 75% RTP.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

REFERENCES

1. FSAR, Chapter ~~[~~15~~]~~.
2. ~~FSAR, Chapter [6]~~. NUREG 0800, Section 4.2, Appendix B
3. Technical Requirements Manual
4. ~~3~~. 10 CFR 50, Appendix A.
5. ~~4~~. 10 CFR 50.

B 3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

B 3.2.~~4~~ AZIMUTHAL POWER TILT (T_q) ~~(Analog)~~

BASES

BACKGROUND

The purpose of this LCO (AZIMUTHAL POWER TILT (T_q)) is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution,

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution, and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Ref~~s~~. 1 ~~and 2~~), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits for linear heat rate (LHR) and departure from nucleate boiling (DNB).

BASES

BACKGROUND (continued)

LINEAR HEAT RATE ( ) AXIAL SHAPE INDEX ( )

The limits on LHR, ~~Total Planar Radial Peaking Factor (F_{XY}^T),~~ Total Integrated Radial Peaking Factor (F_r^T), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

4
1

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LCO limits are not exceeded. The Excore Detector Monitoring System performs this function by continuously monitoring ASI with OPERABLE quadrant symmetric excore neutron detectors and by verifying ASI is maintained within the limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following assumptions are made:

a. The CEA insertion limits of LCO 3.1.5, "Shutdown CEA Insertion Limits," and LCO 3.1.6, "Regulating CEA Insertion Limits," are satisfied,

b. The T_q restrictions of LCO 3.2.~~4~~ 3 are satisfied, and

4
4

c. ~~F_{xy}^T~~ F_r^T does not exceed the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms ~~that have been~~ established for the individual incore detector segments ensure that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances~~, set in conservative directions, for:~~ described in Reference 3.

1

3

~~a. A measurement calculational uncertainty factor of 1.062,~~

~~b. An engineering uncertainty factor of 1.03,~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion, and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02.~~

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or AOOs (Condition 2) (Ref. ~~3~~, GDC 10). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. ~~4~~),

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. ~~3~~, GDC 10),

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 2~~8~~0 cal/gm (Ref. ~~[~~ ~~]~~), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. ~~3~~, GDC 26).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. ~~4~~). This process is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analysis (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate ~~(LHGR)~~ so that the peak cladding temperature does not exceed 2200°F (Ref. ~~4~~). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI~~,~~ F_{xy}^T~~,~~ and F_r^T limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analyses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This potential for fuel cladding damage exists because

Fuel cladding damage does not normally occur while the reactor is operating at conditions outside these LCOs during ~~otherwise~~ normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. Changes in the power distribution cause increased power peaking and correspondingly increased local LHRs.

The T_q satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

certain

The power distribution LCO limits are based on correlations between power peaking and ~~the~~ measured variables used as inputs to the LHR and DNB operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limits on LHR ensure that in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1 with THERMAL POWER > 50% RTP, T_q must be maintained within the limits assumed in accident analysis to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on T_q.

ACTIONS

A.1 and A.2

≤

within 2 hours

If the measured T_q is > ~~[~~0.03~~]~~ and ~~<~~ 0.10, the calculation of T_q may be nonconservative. T_q must be restored within 2 hours or F^T_{XY} ~~and~~ F^T_r must be determined to be within the limits of LCO 3.2.2 ~~and LCO 3.2.3~~, and determined to be within these limits every 8 hours thereafter, as long as T_q is out of limits. Two hours is sufficient time to allow the operator to reposition CEAs, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in F^T_{XY} ~~and~~ F^T_r can be identified before the limits of LCO 3.2.2 ~~and LCO 3.2.3, respectively,~~ are exceeded.

C

~~B~~.1

conservative protection from increased peaking due to potential xenon redistribution and

or B

If Required Actions and associated Completion Times of Condition A are not met, THERMAL POWER must be reduced to ≤ 50% RTP. This requirement provides reasonable assurance that the core is operating within its thermal limits ~~and places the core in a conservative condition~~. Four hours is a reasonable time to reach 50% RTP in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

~~C~~.1, ~~C.2,~~ and ~~C.3~~

With T_q > 0.10, F^T_{XY} ~~and~~ F^T_r must be within ~~their~~ specified limits to ensure that acceptable flux peaking factors are maintained. Based on operating experience, 1 hour is sufficient time for the operator to evaluate these factors. If F^T_{XY} ~~and~~ F^T_r ~~are~~ within limits, operation may proceed for a total of 2 hours after the Condition is entered while attempts are made to restore T_q to within its limit.

If T_q ≤ 0.10 cannot be achieved, power must be reduced to ≤ 50% RTP within ~~2~~ hours. ~~If the tilt is generated due to a CEA misalignment, operating at ≤ 50% RTP allows for the recovery of the CEA. Except as a result of CEA misalignment, T_q > 0.10 is not expected; if it occurs, continued operation of the reactor may be necessary to discover the cause of the tilt. If this procedure is followed, operation is restricted to only those conditions required to identify the cause of the tilt. It is necessary to account explicitly for power asymmetries because the radial power peaking factors used in core power distribution calculations are based on an untilted power distribution.~~

~~If T_q is not restored to within its limits,~~ the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation that causes increased LHRs when the xenon redistributes. ~~If T_q cannot be restored to within its limits within 2 hours, reactor power must be reduced.~~ Reducing THERMAL POWER to ≤ 50% RTP within ~~2~~ hours provides conservative protection from increased peaking due to potential xenon redistribution. ~~The Required Actions are modified by a Note that requires all subsequent actions to be performed once power reduction commences after entering the Condition if T_q is not restored to < 0.10. This procedure ensures corrective action is taken before unrestricted power operation resumes. Following THERMAL POWER reduction to ≤ 50% RTP, T_q must be restored to ≤ [0.03] before THERMAL POWER is increased (Required Action C.3). This Required Action prevents the operator from increasing THERMAL POWER above the conservative limit when the Condition, T_q outside its limits, has existed but allows the unit to continue operation for diagnostic purposes. The Completion Time of Required Action C.3 is modified with a Note to indicate that the cause of the out of limit condition must be corrected prior to increasing THERMAL POWER. This Note also indicates that subsequent power operation above 50% RTP may proceed provided that the measured T_q is verified ≤ [0.03] at least once per hour for 12 hours, or until verified at 95% RTP. This ensures that the power distribution is responding as predicted. The Completion Time of 12 hours is a historical value that allows an acceptable exit from the LCO after the T_q value is verified acceptable for 12 hours or until 95% RTP is reached.~~

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.~~4~~.1

~~[T_q must be calculated at 12 hour intervals. The 12 hour Frequency prevents significant xenon redistribution between Surveillances.~~

Tq must be determined at 12 hour intervals when one excore detector is inoperable and > 75% RTP.

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

REFERENCES

1. FSAR, Chapter ~~[~~15~~]~~.
2. ~~FSAR, Chapter [6]~~. NUREG 0800, Section 4.2, Appendix B
3. Technical Requirements Manual
4. ~~3~~. 10 CFR 50, Appendix A.
5. ~~4~~. 10 CFR 50.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.3, BASES, AZIMUTHAL POWER TILT (T_q)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS contains tolerances to be applied to the incore detector alarm setpoints. PSL Unit 1 UFSAR, Section 13.8, Licensee-Controlled Technical Specification Requirements, and PSL Unit 2 UFSAR, Section 13.7, Licensee-Controlled Technical Specification Requirements, each contain these tolerances to be applied to the incore detector alarm setpoints. Additionally, these tolerances will be provided in the Technical Requirements Manual. Therefore, the tolerances to be applied to the incore detector alarm setpoints are deleted.

4. The ISTS includes ISTS 3.2.1, Linear Heat Rate (LHR), ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^T), ISTS 3.2.3, Total Integrated Radial Peaking Factor (F_r^T), ISTS 3.2.4, Azimuthal Power Tilt (T_q), and ISTS 3.2.5, Axial Shape Index (ASI). CTS do not include a Specification for ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^T). The CTS and ISTS are renumbered. CTS 3.2.3, Total Integrated Radial Peaking Factor (F_r^T), CTS 3.2.4, Azimuthal Power Tilt (T_q), and CTS 3.2.5, Axial Shape Index (ASI), are renumbered as ITS 3.2.2, ITS 3.2.3, and ITS 3.2.4, respectively.

5. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

6. ISTS Condition B is renumbered as ITS Condition C, and ISTS Condition C is renumbered as ITS Condition B. ITS Condition C deletes the verbiage “of Condition A” so ITS Condition C and Required Action C.1 apply to both ITS Condition A and ITS Condition B.

7. ISTS Bases Actions C.1, C.2, and C.3 changes support the CTS 3.2.4 and ITS 3.2.3 changes.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.3, AZIMUTHAL POWER TILT (T_q)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 4

ITS 3.2.4, AXIAL SHAPE INDEX (ASI)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

POWER DISTRIBUTION LIMITS

~~DNB PARAMETERS~~ AXIAL SHAPE INDEX (ASI)

LIMITING CONDITION FOR OPERATION

4

3.2.~~5~~ The following DNB related parameters shall be maintained within the limits:

See ITS 3.4.1

a. Cold Leg Temperature as shown on Table 3.2-1 of the COLR,

b. Pressurizer Pressure* as shown on Table 3.2-1 of the COLR,

c. Reactor Coolant System Total Flow Rate - greater than or equal to 375,000 gpm, and

LCO 3.2.4

~~d. AXIAL SHAPE INDEX as shown on Figure 3.2-4 of~~ the COLR.

The ASI shall be maintained within the limits specified in

Applicability

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 40% RTP.

L01

ACTION:

ASI not within limits

ASI

s

Required Action A.1 ~~With any of the above parameters~~ exceeding ~~its limit~~, restore ~~the parameter~~ to within ~~its~~ limit

Required Action B.1 within 2 hours or reduce THERMAL POWER to ~~≤~~ 5% ~~of RATED THERMAL POWER~~ within ~~the next~~ 4 hours.

RTP

< 40

L01

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (ASI)

Verify ASI is

specified in the COLR

~~4.2.5.1~~ ~~Each of the DNB related parameters shall be verified to be~~ within ~~their~~ limits ~~by instrument readout~~ in accordance with the Surveillance Frequency Control Program.

See ITS 3.4.1

~~4.2.5.2~~ The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement** in accordance with the Surveillance Frequency Control Program.

See ITS 3.4.1

---

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% per minute of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

** Not required to be performed until THERMAL POWER is ≥ 90% of RATED THERMAL POWER.

~~Relocated to the COLR~~

POWER DISTRIBUTION LIMITS

~~DNB PARAMETERS~~ AXIAL SHAPE INDEX (ASI)

LIMITING CONDITION FOR OPERATION

3.2.~~5~~ 4

The following DNB-related parameters shall be maintained within the limits:

a. Cold Leg Temperature as shown on Table 3.2-2 of the COLR,

b. Pressurizer Pressure* as shown on Table 3.2-2 of the COLR,

c. Reactor Coolant System Total Flow Rate - greater than or equal to 375,000 gpm, and

See ITS 3.4.1

LCO 3.2.4

~~d. AXIAL SHAPE INDEX as shown on Figure 3.2-4 of~~ the COLR.

The ASI shall be maintained within the limits specified in

Applicability

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 40% RTP.

L01

ACTION:

ASI not within limits

Required Action A.1 ~~With any of the above parameters exceeding its limit~~, restore ~~the parameter~~ to within its limit within

Required Action B.1 2 hours or reduce THERMAL POWER to ~~≤~~ 5% ~~of RATED THERMAL POWER~~ within ~~the next~~ 4 hours.

ASI

RTP

s

< 40

L01

SR 3.2.4.1 (ASI)

SURVEILLANCE REQUIREMENTS

Verify ASI is

specified in the COLR

~~4.2.5.1~~ ~~Each of the DNB-related parameters shall be verified to be~~ within ~~their~~ limits ~~by instrument readout~~ in accordance with the Surveillance Frequency Control Program.

See ITS 3.4.1

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement** in accordance with the Surveillance Frequency Control Program.

See ITS 3.4.1

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* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% per minute of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

** Not required to be performed until THERMAL POWER is ≥ 90% of RATED THERMAL POWER.

A01

~~DELETED~~

DISCUSSION OF CHANGES
ITS 3.2.4, AXIAL SHAPE INDEX (ASI)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.2.5 states that the Applicability is MODE 1 and when the LCO is not met, the actions require restoring the parameter within 2 hours or reducing THERMAL POWER to ≤ 5% of RATED THERMAL POWER (i.e, MODE 2) within the next 4 hours. ITS 3.2.4 states that the Applicability is MODE 1 with THERMAL POWER ≥ 40% RTP and the Required Actions are consistent with CTS except THERMAL is reduced to < 40% RTP, which results placing the unit in a condition where ASI no longer applies. This changes the CTS by relaxing the Applicability to MODE 1 with THERMAL POWER ≥ 40% RTP and corresponding change to only require THERMAL POWER to be reduced < 40% RTP consistent with the revised Applicability.

The purpose of CTS 3.2.5 Applicability and ITS 3.2.4 Applicability is to establish the plant conditions where ASI is assumed in the safety analysis. PSL Unit 1 and Unit 2 safety analysis apply ASI at ≥ 40% RTP. The ASI limits curve specified in the COLR also explicitly state that the limits do not apply below 40% RTP. Therefore, below 40% RTP, the ASI power distribution limit is not applicable. Other limits assumed in the accident analysis ensure fuel damage does not occur below 40% RTP (e.g., linear heat rate (LHR) limit, power dependent rod insertion limits, CEA alignment, sequence, and overlap limits, and reactor coolant conditions; pressure, temperature, flow). The power distribution limits are based on correlations between power peaking and certain measured variables used as

inputs to the LHR and DNB operating limits. The limitation on ASI, along with the limitations of LCO 3.3.1, "Reactor Protection System Instrumentation," represents a conservative envelope of operating conditions consistent with the assumptions that have been analytically demonstrated adequate for maintaining an acceptable minimum DNBR throughout all AOOs.

This change is acceptable because the LCO requirements applied in the ITS Applicability continues to ensure process variables, structures, systems, and components are maintained in the conditions assumed in the safety analyses and licensing basis. In MODE 1 with THERMAL POWER ≥ 40% RTP, ASI must be maintained within the power distribution limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER < 40% RTP, THERMAL POWER is not sufficient to require a limit on ASI power distribution.

This change is designated as less restrictive because the LCO requirements will not be applied under plant conditions in which they would be applied under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

3.2.~~5~~ AXIAL SHAPE INDEX (ASI) ~~(Analog)~~

4

4
3

LCO 3.2.5 LCO 3.2.~~5~~ The ASI shall be maintained within the limits specified in the COLR.

4

Applicability DOC L01 APPLICABILITY: MODE 1 with THERMAL POWER ~~> 20~~% RTP.

≥ 40

5

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action A. ASI not within limits.	A.1 Restore ASI to within limits.	2 hours
Action DOC L01 B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ~~≤ 20~~% RTP. < 40	4 hours

5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.2.5.1 SR 3.2.~~5~~.1 Verify ASI is within limits specified in the COLR. 4	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

3

2

3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

4
3

3.2.~~5~~ AXIAL SHAPE INDEX (ASI) ~~(Analog)~~

4

LCO 3.2.5

LCO 3.2.~~5~~ The ASI shall be maintained within the limits specified in the COLR.

4

Applicability
DOC L01

APPLICABILITY: MODE 1 with THERMAL POWER ~~> 20~~% RTP.

≥ 40

5

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action A. ASI not within limits.	A.1 Restore ASI to within limits.	2 hours
Action DOC L01 B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ~~≤ 20~~% RTP. < 40	4 hours

5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.2.5.1 SR 3.2.~~5~~.1 Verify ASI is within limits specified in the COLR. 4	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

3

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.4, AXIAL SHAPE INDEX (ASI)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS includes ISTS 3.2.1, Linear Heat Rate (LHR), ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}), ISTS 3.2.3, Total Integrated Radial Peaking Factor (F_{r}^{T}), ISTS 3.2.4, Azimuthal Power Tilt (T_q), and ISTS 3.2.5, Axial Shape Index (ASI). CTS do not include a Specification for ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}). The CTS and ISTS are renumbered. CTS 3.2.3, Total Integrated Radial Peaking Factor (F_{r}^{T}), CTS 3.2.4, Azimuthal Power Tilt (T_q), and CTS 3.2.5, Axial Shape Index (ASI), are renumbered as ITS 3.2.2, ITS 3.2.3, and ITS 3.2.4, respectively.

4. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

5. ISTS 3.2.5 (renumbered as ITS 3.2.4) Applicability and Required Action B.1 are changed to reflect relaxing the CTS Applicability. ITS 3.2.4 Applicability is MODE 1 with THERMAL POWER ≥ 40% RTP. This changes the CTS by relaxing the Applicability from MODE 1, to MODE 1 with THERMAL POWER ≥ 40% RTP. See DOC L01 for a discussion of changes for relaxing the CTS Applicability.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

5

B 3.2.~~5~~ AXIAL SHAPE INDEX (ASI) ~~(Analog)~~

4

4

BASES

BACKGROUND

The purpose of this LCO (AXIAL SHAPE INDEX (ASI)) is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution,

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution, and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Ref~~s~~. 1 ~~and 2~~), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

1

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on linear heat rate (LHR) and departure from nucleate boiling (DNB).

BASES

BACKGROUND (continued)

The limits on LHR, ~~Total Planar Radial Peaking Factor (F_{XY}^T),~~ Total Integrated Radial Peaking Factor (F_r^T), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and ~~are~~ is capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring ~~the~~ ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following conditions are assumed:

a. The CEA insertion limits of LCO 3.1.5, "Shutdown CEA Insertion Limits," and LCO 3.1.6, "Regulating CEA Insertion Limits," are satisfied,

b. The T_q restrictions of LCO 3.2.~~4~~ are satisfied, and

c. ~~F_{xy}^T~~ F_r^T does not exceed the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms ~~that have been~~ established for the individual incore detector segments ensure that the peak LHR is maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances~~, set in conservative directions, as follows:~~ described in Reference 3.

~~a. A measurement calculational uncertainty factor of 1.062,~~

~~b. An engineering uncertainty factor of 1.03,~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion, and~~

~~d. A THERMAL POWER measurement uncertainty factor of 1.02.~~

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or AOOs (Condition 2) (Ref. ~~3~~, GDC 10). The power distribution and CEA insertion and alignment LCOs prevent core power distributions ~~from reaching levels~~ that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. ~~4~~),

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. ~~3~~, GDC 10),

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 2~~8~~0 cal/gm (Ref. ~~[ ]~~), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. ~~3~~, GDC 26).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. ~~4~~). This limitation is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations ~~among~~ between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. ~~4~~). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI~~,~~ F_{XY}^T, and F_r^T limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not normally occur while the reactor is operating at conditions outside these LCOs during normal operation. Fuel cladding damage results, however, when an accident or AOO occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

The ASI satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB operating limits. These power distribution LCO limits, except T_q, are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

The limitation on ASI, along with the limitations of LCO 3.3.1, "Reactor Protection System Instrumentation," represents a conservative envelope of operating conditions consistent with the assumptions that have been analytically demonstrated adequate for maintaining an acceptable minimum DNBR throughout all AOOs. Of these, the loss of flow transient is the most limiting. Operation of the core with conditions within the specified limits ensures that an acceptable minimum margin from DNB conditions is maintained in the event of any AOO, including a loss of flow transient.

APPLICABILITY

In MODE 1 with THERMAL POWER ~~>~~ ≥ 40 ~~20~~% RTP, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on the core power distribution. ~~Below 20% RTP the incore detector accuracy is not reliable~~. 6

ACTIONS

A.1

Operating the core within ASI limits specified in the COLR and within the limits of LCO 3.3.1 ensures an acceptable margin for DNB and for maintaining local power density in the event of an AOO. Maintaining ASI within limits also ensures that the limits of 10 CFR 50.46 are not exceeded during accidents. The Required Actions to restore ASI must be completed within 2 hours to limit the duration the plant is operated outside the initial conditions assumed in the accident analyses. In addition, this Completion Time is sufficiently short that the xenon distribution in the core cannot change significantly.

BASES

ACTIONS (continued)

B.1

If the ASI cannot be restored to within its specified limits~~, or ASI cannot be determined because of Excore Detector Monitoring System inoperability,~~ core power must be reduced. Reducing THERMAL POWER to ~~≤~~ 20% RTP provides reasonable assurance that the core is operating farther from thermal limits and places the core in a conservative condition. Four hours is a reasonable amount of time, based on operating experience, to reduce THERMAL POWER to ~~≤ 20~~% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.~~5~~.1

Verifying that the ASI is within the specified limits provides reasonable assurance that the core is not approaching DNB conditions. ~~[A Frequency of 12 hours is adequate for the operator to identify trends in conditions that result in an approach to the ASI limits, because the mechanisms that affect the ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause the ASI to change slowly and should be discovered before the limits are exceeded.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

REFERENCES

1. FSAR, Chapter ~~[~~15~~]~~.
2. ~~FSAR, Chapter [6].~~ NUREG 0800, Section 4.2, Appendix B
3. Technical Requirements Manual
4. ~~3.~~ 10 CFR 50, Appendix A.
5. ~~4.~~ 10 CFR 50.46.

B 3.2 POWER DISTRIBUTION LIMITS ~~(Analog)~~

B 3.2.~~5~~ AXIAL SHAPE INDEX (ASI) ~~(Analog)~~

4

BASES

BACKGROUND

The purpose of this LCO (AXIAL SHAPE INDEX (ASI)) is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution,

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution, and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Ref~~s~~. 1 ~~and 2~~), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameter rs, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on linear heat rate (LHR) and departure from nucleate boiling (DNB).

BASES

BACKGROUND (continued)

The limits on LHR, ~~Total Planar Radial Peaking Factor (F_{XY}^T),~~ Total Integrated Radial Peaking Factor (F_r^T), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and ~~are~~ is capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring ~~the~~ ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following conditions are assumed:

a. The CEA insertion limits of LCO 3.1.5, "Shutdown CEA Insertion Limits," and LCO 3.1.6, "Regulating CEA Insertion Limits," are satisfied,

b. The T_q restrictions of LCO 3.2.~~4~~ 3 are satisfied, and

c. ~~F_{xy}^T~~ F_r^T does not exceed the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms ~~that have been~~ established for the individual incore detector segments ensure that the peak LHR is maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances~~, set in conservative directions, as follows:~~ described in Reference 3.

~~a. A measurement calculational uncertainty factor of 1.062,~~

~~b. An engineering uncertainty factor of 1.03,~~

~~c. An allowance of 1.002 for axial fuel densification and thermal expansion, and~~

~~A THERMAL POWER measurement uncertainty factor of 1.02.~~

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or AOOs (Condition 2) (Ref. ~~3~~, GDC 10). The power distribution and CEA insertion and alignment LCOs prevent core power distributions ~~from reaching levels~~ that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. ~~4~~),

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. ~~3~~, GDC 10),

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 2~~8~~0 cal/gm (Ref. ~~[~~ ~~]~~), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. ~~3~~, GDC 26).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. ~~4~~). This limitation is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations ~~among~~ between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. ~~4~~). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI~~,~~ F_{XY}^T~~,~~ and F_r^T limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not normally occur while the reactor is operating at conditions outside these LCOs during normal operation. Fuel cladding damage results, however, when an accident or AOO occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

The ASI satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB operating limits. These power distribution LCO limits, except T_q, are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

The limitation on ASI, along with the limitations of LCO 3.3.1, "Reactor Protection System Instrumentation," represents a conservative envelope of operating conditions consistent with the assumptions that have been analytically demonstrated adequate for maintaining an acceptable minimum DNBR throughout all AOOs. Of these, the loss of flow transient is the most limiting. Operation of the core with conditions within the specified limits ensures that an acceptable minimum margin from DNB conditions is maintained in the event of any AOO, including a loss of flow transient.

APPLICABILITY

In MODE 1 with THERMAL POWER ≥ 40 ~~> 20~~% RTP, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on the core power distribution. ~~Below 20% RTP the incore detector accuracy is not reliable~~. 6

ACTIONS

A.1

Operating the core within ASI limits specified in the COLR and within the limits of LCO 3.3.1 ensures an acceptable margin for DNB and for maintaining local power density in the event of an AOO. Maintaining ASI within limits also ensures that the limits of 10 CFR 50.46 are not exceeded during accidents. The Required Actions to restore ASI must be completed within 2 hours to limit the duration the plant is operated outside the initial conditions assumed in the accident analyses. In addition, this Completion Time is sufficiently short that the xenon distribution in the core cannot change significantly.

BASES

ACTIONS (continued)

B.1

If the ASI cannot be restored to within its specified limits~~, or ASI cannot be determined because of Excore Detector Monitoring System inoperability,~~ core power must be reduced. Reducing THERMAL POWER to ~~≤~~ 20% RTP provides reasonable assurance that the core is operating farther from thermal limits and places the core in a conservative condition. Four hours is a reasonable amount of time, based on operating experience, to reduce THERMAL POWER to ~~≤~~ 20% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.~~5~~.1

Verifying that the ASI is within the specified limits provides reasonable assurance that the core is not approaching DNB conditions. ~~[A Frequency of 12 hours is adequate for the operator to identify trends in conditions that result in an approach to the ASI limits, because the mechanisms that affect the ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause the ASI to change slowly and should be discovered before the limits are exceeded.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

REFERENCES

1. FSAR, Chapter ~~[~~15~~]~~.
2. ~~FSAR, Chapter [6].~~ NUREG 0800, Section 4.2, Appendix B
3. Technical Requirements Manual
4. ~~3.~~ 10 CFR 50, Appendix A.
5. ~~4.~~ 10 CFR 50.46.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.4, BASES, AXIAL SHAPE INDEX (ASI)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS contains tolerances to be applied to the incore detector alarm setpoints. PSL Unit 1 UFSAR, Section 13.8, Licensee-Controlled Technical Specification Requirements, and PSL Unit 2 UFSAR, Section 13.7, Licensee-Controlled Technical Specification Requirements, each contain these tolerances to be applied to the incore detector alarm setpoints. Additionally, these tolerances will be provided in the Technical Requirements Manual. Therefore, the tolerances to be applied to the incore detector alarm setpoints are deleted.

4. The ISTS includes ISTS 3.2.1, Linear Heat Rate (LHR), ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}), ISTS 3.2.3, Total Integrated Radial Peaking Factor (F_{r}^{T}), ISTS 3.2.4, Azimuthal Power Tilt (T_q), and ISTS 3.2.5, Axial Shape Index (ASI). CTS do not include a Specification for ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}). The CTS and ISTS are renumbered. CTS 3.2.3, Total Integrated Radial Peaking Factor (F_{r}^{T}), CTS 3.2.4, Azimuthal Power Tilt (T_q), and CTS 3.2.5, Axial Shape Index (ASI), are renumbered as ITS 3.2.2, ITS 3.2.3, and ITS 3.2.4, respectively.

5. The type of plant (Analog) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

6. ISTS 3.2.5 Bases (renumbered as ITS 3.2.4) Applicability discussion and Action B.1 discussion are changed to reflect relaxing the CTS Applicability. ITS 3.2.4 Applicability is MODE 1 with THERMAL POWER ≥ 40% RTP. This changes the CTS by relaxing the Applicability from MODE 1, to MODE 1 with THERMAL POWER ≥ 40% RTP. See DOC L01 for a discussion of changes for relaxing the CTS Applicability. Additionally, in the Applicability discussion, the statement “below 20% RTP the incore detector accuracy is not reliable” is deleted because the state of the incore detector accuracy at 20% RTP it is not relevant to the condition that below 40% RTP the ASI LCO no longer applies. In the Action B.1 discussion, the statement “or ASI cannot be determined because of Excore Detector Monitoring System inoperability” is deleted because the operability of the Excore Detector Monitoring System is not relevant to the Action B.1 action that when ASI is not restored to within limits within 2 hours, then THERMAL POWER must be reduced to below 40% RTP within 4 hours.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.4, AXIAL SHAPE INDEX (ASI)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

ISTS Not Adopted

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.2 POWER DISTRIBUTION LIMITS (Analog)

3.2.2 Total Planar Radial Peaking Factor (F_{xy}^{T}) (Analog)

LCO 3.2.2 The calculated value of F_{xy}^{T} shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Required Actions shall be completed if this Condition is entered. ---------- F_{xy}^{T} not within limits.	A.1 Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_{xy}^{T} to within the limits specified in the COLR.	6 hours
	AND	
	A.2 Withdraw the control element assemblies (CEAs) to or beyond the long term steady state insertion limits of LCO 3.1.6, "Regulating CEAs," as specified in the COLR.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	NOTE SR 3.2.2.2 and SR 3.2.2.3 shall be completed each time SR 3.2.2.1 is required. F^T_{xy} shall be determined by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the long term steady state insertion limit, as specified in the COLR.	
	Verify the value of F^T_{xy}	Once prior to operation above 70% RTP after each fuel loading AND [Each 31 days of accumulated operation in MODE 1 OR In accordance with the Surveillance Frequency Control Program]
SR 3.2.2.2	Verify the value of F_{xy}.	In accordance with the Frequency requirements of SR 3.2.2.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.2.3 Verify the value of T_q.	In accordance with the Frequency requirements of SR 3.2.2.1

JUSTIFICATION FOR DEVIATIONS
ISTS 3.2.2, TOTAL PLANAR RADIAL PEAKING FACTOR (F_{xy}^{T}) (ANALOG)

1. ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}) (Analog) Specification is not included in the St. Lucie Plant (PSL) Unit 1 and Unit 2 ITS because the PSL CTS do not include a Specification for Total Planar Radial Peaking Factor (F_{xy}^{T}) (Analog).

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.2 POWER DISTRIBUTION LIMITS (Analog)

B 3.2.2 Total Planar Radial Peaking Factor (F^{T}_{XY}) (Analog)

BASES

BACKGROUND The purpose of this LCO (Total Planar Radial Peaking Factor (F^{T}_{XY})) is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO decreases or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution,

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution, and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings (LSSS) and this LCO are based on accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the linear heat rate (LHR) and departure from nucleate boiling (DNB).

BASES

BACKGROUND (continued)

The limits on LHR, F^T_{XY}, Total Integrated Radial Peaking Factor (F^T_r), T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following assumptions are made:

a. The CEA insertion limits of LCO 3.1.5, "Shutdown CEA Insertion Limits," and LCO 3.1.6, "Regulating CEA Insertion Limits," are satisfied,

b. The T_q restrictions of LCO 3.2.4 are satisfied, and

c. F^T_{XY} does not exceed the limits of this LCO.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms that have been established for the individual incore detector segments ensure that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, for:

a. A measurement calculational uncertainty factor of 1.062,

b. An engineering uncertainty factor of 1.03,

c. An allowance of 1.002 for axial fuel densification and thermal expansion, and

d. A THERMAL POWER measurement uncertainty factor of 1.02.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or AOOs (Condition 2) (Ref. 3, GDC 10). The Power Distribution and CEA Insertion and Alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 4),

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 3, GDC 10),

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. [ ]), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck, fully withdrawn (Ref. 3, GDC 26).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This limiting is accomplished by maintaining the power distribution and reactor coolant conditions such that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and the uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 4). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Reactor Coolant System ensure that these criteria are met as long as the core is operated within the ASI, F^{T}_{XY}, F^{T}_{r}, and T_q limits specified in the COLR. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding damage does not normally occur while at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, should an accident or AOO occur from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHR.

F^{T}_{XY} satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO — The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB ratio operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY — In MODE 1, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS — A.1 and A.2

A Note modifies Condition A to require Required Actions A.1 and A.2 to be completed if the Condition is entered. This ensures that corrective action is taken prior to unrestricted operation.

The limitations on F^{T}_{XY} provided in the COLR ensure that the assumptions used in the analysis for establishing the LHR, LCO, and LSSS remain valid during operation at the various allowable CEA group insertion limits. If F^{T}_{XY} exceeds its basic limitation, operation may continue under the additional restrictions imposed by these Required Actions (reducing THERMAL POWER and withdrawing CEAs to or beyond the long term steady state insertion limits of LCO 3.1.6), because these additional restrictions adequately ensure that the assumptions used in establishing the LHR, LCO, and LSSS remain valid (Ref. 3). Six hours to return F^{T}_{XY} to within its limit is reasonable and ensures that all CEAs meet the long term steady state insertion limits of LCO 3.1.6.

BASES

ACTIONS (continued)

B.1

If F^{T}_{r} cannot be returned to within its limit, THERMAL POWER must be reduced. A change to MODE 2 provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The periodic Surveillance to determine the calculated F^{T}_{XY} ensures that F^{T}_{XY} remains within the range assumed in the analysis throughout the fuel cycle. Determining the measured F^{T}_{XY} after each fuel loading prior to the reactor exceeding 70% RTP ensures that the core is properly loaded.

[Performance of the Surveillance every 31 days of accumulated operation in MODE 1 provides reasonable assurance that unacceptable changes in the F^{T}_{XY} are promptly detected.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

The power distribution map can only be obtained after THERMAL POWER exceeds 20% RTP because the incore detectors are not reliable below 20% RTP.

The SR is modified by a Note that requires that SR 3.2.2.2 and SR 3.2.2.3 be completed each time SR 3.2.1.1 is completed. (Values computed by these SRs are required to perform SR 3.2.2.1.) The Note also requires that the incore detectors be used to determine F^{T}_{XY} by using them to obtain a power distribution map with all full length CEAs above the long term steady state insertion limits, as specified in the COLR.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2 and SR 3.2.2.3

Measuring the value of F_{XY} and T_q each time a calculated value of F^{I}_{XY} is required ensures that the calculated value of F^{I}_{XY} accurately reflects the condition of the core.

The Frequency for these Surveillances is in accordance with the Frequency requirements of SR 3.2.2.1, because these SRs provide information to complete SR 3.2.2.1.

REFERENCES
1. FSAR, Chapter [15].
2. FSAR, Chapter [6].
3. 10 CFR 50, Appendix A.
4. 10 CFR 50.46.

JUSTIFICATION FOR DEVIATIONS
ISTS 3.2.2, BASES, TOTAL PLANAR RADIAL PEAKING FACTOR (F_{xy}^{T}) (ANALOG)

1. ISTS 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}) (Analog) Specification is not included in the St. Lucie Plant (PSL) Unit 1 and Unit 2 ITS because the PSL CTS do not include a Specification for Total Planar Radial Peaking Factor (F_{xy}^{T}) (Analog).

ENCLOSURE 2

VOLUME 8

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.3
INSTRUMENTATION

Revision 1

LIST OF ATTACHMENTS

1. 3.3.1, Reactor Protective System (RPS) Instrumentation
2. 3.3.2, Reactor Protective System (RPS) Logic and Trip Initiation
3. 3.3.3, Engineered Safety Features Actuation System (ESFAS) Instrumentation
4. 3.3.4, Engineered Safety Features Actuation System (ESFAS) Logic and Manual Actuation
5. 3.3.5, Diesel Generator (DG) – Loss of Voltage Start (LOVS)
6. 3.3.6, Containment Isolation Instrumentation – Refueling
7. 3.3.7, Control Room Isolation Signal (CRIS)
8. 3.3.8, Fuel Pool Area Radiation Instrumentation
9. 3.3.9, Post Accident Monitoring (PAM) Instrumentation
10. 3.3.10, Remote Shutdown System
11. 3.3.11, Logarithmic Neutron Flux Monitoring
12. ISTS Not Adopted

ATTACHMENT 1

3.3.1, Reactor Protective System (RPS) Instrumentation

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3/4.3 INSTRUMENTATION

SYSTEM (RPS)

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.1

3.3.1.1 ~~As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1~~ shall be OPERABLE.

Four RPS trip units and associated instrument and bypass removal channels for each Function in Table 3.3.1-1

A02

Applicability **APPLICABILITY:** As shown in Table 3.3-1.

Add proposed ACTIONS note.

A03

ACTIONS **ACTION:**

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1
SR 3.3.1.2
SR 3.3.1.3
SR 3.3.1.4
SR 3.3.1.5
SR 3.3.1.6
SR NOTE

4.3.1.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-1.

SR 3.3.1.7

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

SR 3.3.1.8

SR 3.3.1.9

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing. ~~Each test shall include at least one channel per function.~~

SR 3.3.1.9 NOTE

A04

Table 3.3.1-1

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

SYSTEM (RPS) - OPERATING

OR OTHER SPECIFIED CONDITIONS

LA01 LA01 A02

LCO 3.3.1 Applicability

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
	1. Manual Reactor Trip	2	1	2	1, 2 and *	1	See ITS 3.3.2
Function 1 Footnote c	2. Power Level – High (Variable)	4	2(a) (c)	3(f) LA02	1, 2	2	
Function 3 Footnote c	3. Reactor Coolant Flow – Low	4/SG	2(a)/SG (c)	3/SG	1, 2 (e)	2	
Function 4	4. Pressurizer Pressure – High	4	2	3	1, 2	2	
Function 5	5. Containment Pressure – High	4	2	3	1, 2	2	
Function 6 Footnote f	6. Steam Generator Pressure – Low	4/SG	2(b) (f)/SG	3/SG	1, 2	2	
Function 7a and b	7. Steam Generator Water Level – Low	4/SG	2/SG	3/SG	1, 2	2	
Function 8 Footnotes g and h	8. Local Power Density – High	4	2(c) (g)	3	1 (h)	2	L06
Function 9a Footnote c	9. Thermal Margin/Low Pressure	4	2(a) (c)	3	1, 2 (e)	2	
Function 9b Footnote c	9a. Steam Generator Pressure Difference – High	4	2(a) (c)	3	1, 2 (e)	2	
Function 10 Footnotes g and h	10. Loss of ~~Turbine –~~ Hydraulic Fluid Pressure - Low (Load - Turbine)	4	2(c) (g)	3	1 (h)	2	L06

Table 3.3.1-1

TABLE 3.3-1 (Continued)

SYSTEM (RPS)

- OPERATING

REACTOR PROTECTIVE INSTRUMENTATION

OR OTHER SPECIFIED CONDITIONS

A02

LA01

LA01

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
11. Wide Range Logarithmic Neutron Flux Monitor						
a. Startup and Operating -- Rate of Change of Power – High (Power)	4	2(d)	3	1**, 2 and *	2	L01
b. Shutdown	4	0	2	3, 4, 5	3	See ITS 3.3.11
12. Reactor Protection System Logic	4	2	4	1, 2*	4	See ITS 3.3.2
13. Reactor Trip Breakers	4	2	4	1, 2*	4	

Function 2 Footnotes d and e

TABLE 3.3-1 (Continued)

TABLE NOTATION

L01

~~* With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.~~

A01

Trip not applicable when Thermal Power is > 15% RTP.

Footnote e

** ~~Mode 1 applicable only when Power Range Neutron Flux power ≤ 15% of RATED THERMAL POWER.~~

Footnote c

(a) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is ≥ 1% of RATED THERMAL POWER.

Footnote f

(b) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

L06

is not applicable and

Footnotes g & h

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is ≥ 15% of RATED THERMAL POWER.

Footnote d

(d) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is ≥ 10^{-4}% and Power Range Neutron Flux power ≤ 15% of RATED THERMAL POWER.

~~(e) Deleted.~~

LA02

~~(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.~~

L06

h. Trip is only applicable in MODE 1 ≥ 15% RTP.

ACTION STATEMENTS

See ITS 3.3.2

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION A

ACTION 2 - ~~With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:~~

Place affected trip unit

a. ~~The inoperable channel is placed~~ in ~~either the~~ bypass~~ed~~ or trip~~ped condition~~ within 1 hour. ~~For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition~~.

One or more Functions with one RPS trip unit or associated instrument channel inoperable except for Condition C (excore channel not calibrated with incore detectors).

AND
Restore channel to Operable status within 48 hours.
OR
Place affected unit in trip within 48 hours.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

~~b. Within one hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.~~

A07

ACTION A

~~c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on than channel provided the other inoperable channel is placed in the tripped condition.~~

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

See ITS 3.3.11

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

See ITS 3.3.2

Add proposed ACTIONS B and C

L02

Add proposed ACTIONS D and E.

L03

Add proposed ACTIONS G and F

A08

~~DELETED~~

ITS

A01

ITS 3.3.1

Table 3.3.1-1

TABLE 4.3-1
REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	SR 3.3.1.1 CHANNEL CHECK	SR 3.3.1.2, SR 3.3.1.3, SR 3.3.1.5, SR 3.3.1.8 CHANNEL CALIBRATION (a)(b)	SR 3.3.1.4, SR 3.3.1.6, SR 3.3.1.7 CHANNEL FUNCTIONAL TEST (a)(b)	MODES IN WHICH SURVEILLANCE REQUIRED (M01, M02)	
	1. Manual Reactor Trip	N/A	N.A.	S/U(1)	N/A	See ITS 3.3.2
Function 1	2. Power Level – High (Variable)					
	a. Nuclear Power	SFCP	SFCP(2), SFCP(3), SFCP(5)	SFCP	1,2	
	b. ΔT Power	SFCP	SFCP(4), SFCP	SFCP	1	
Function 3	3. Reactor Coolant Flow – Low	SFCP	SFCP	SFCP	1, 2	
Function 4	4. Pressurizer Pressure – High	SFCP	SFCP	SFCP	1, 2	
Function 5	5. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2	
Function 6	6. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2	
Functions 7a and 7b	7. Steam Generator Water Level – Low	SFCP	SFCP	SFCP~~(6, 7)~~	1, 2	
Function 8	8. Local Power Density – High	SFCP	SFCP	SFCP	1	
Function 9a	9. Thermal Margin/Low Pressure	SFCP	SFCP	SFCP	1, 2	
Function 9b	9a. Steam Generator Pressure Difference – High	SFCP	SFCP	SFCP	1, 2	
Function 10	10. Loss of ~~Turbine~~ Hydraulic Fluid Pressure – Low (Load - Turbine)	N.A.	M03 ~~N.A.~~ (SFCP)	S/U(1)	N.A.	
Function 2	11. Wide Range Logarithmic Neutron Flux Monitor (Power Rate of Change - High)	SFCP	M03 ~~N.A.~~ (SFCP)	S/U(1)	1, 2, 3, 4, 5 ~~and *~~	See ITS 3.3.11
	12. Reactor Protection System Logic	N.A.	N.A.	SFCP and S/U(1)	1, 2 and *	L01
	13. Reactor Trip Breakers	N.A.	N.A.	SFCP	1, 2 and *	See ITS 3.3.2

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - ~~With reactor trip breaker closed.~~ L01

SR 3.3.1.6 (1) - If not performed in previous 7 days.

Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. M04

SR 3.3.1.2 NOTE 1 (2) - ~~Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Pwr – ΔT Pwr."~~ During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

SR 3.3.1.2 NOTE 2

Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. M05

SR 3.3.1.3 and NOTE Required Action C.2 (3) - ~~Above 15% of RATED THERMAL POWER,~~ recalibrate the excore detectors ~~which monitor the AXIAL SHAPE INDEX by~~ using the incore detectors or restrict THERMAL POWER ~~during subsequent operations~~ to ≤ 90% of the maximum allowed THERMAL POWER level ~~with the existing Reactor Coolant Pump combination~~. LA03 A05

SR 3.3.1.2 (4) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

SR 3.3.1.5 NOTE and SR 3.3.1.8 NOTE (5) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

instrument channel

Table 3.3.1-1 Footnote a (6) - If the as-found setpoint is ~~either~~ outside its predefined as-found acceptance criteria band ~~or is not conservative with respect to the Allowable Value~~, then the channel shall be ~~declared inoperable and shall be~~ evaluated to verify that it is functioning as required before returning the channel to service. M01

Table 3.3.1-1 Footnote b (7) - ~~The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Field Trip Setpoint, otherwise that channel shall not be returned to OPERABLE status. The Field Trip Setpoint and the methodology used to determine the Field Trip Setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in the UFSAR Section 7.2.~~ M02

The instrument channel setpoint shall be reset to a value that is within the as-left acceptance criteria band around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left acceptance criteria bands apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP and the methodologies used to determine the as-found and as-left acceptance criteria bands are specified in the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively.

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

LCO 3.3.1 2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the ~~Trip Setpoint~~ Allowable values shown in Table 2.2-1. LA04

APPLICABILITY: AS SHOWN FOR EACH CHANNEL IN TABLE 3.3-1.

ACTION:

ACTION A With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the ~~Trip Setpoint~~ Allowable value. LA04

Table 3.3.1-1

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

LA04

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
	1. Manual Reactor Trip	Not Applicable	Not Applicable	
Function 1	2. Power Level – High (1) (c) Variable Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.	A06
Function 3	3. Reactor Coolant Flow – Low (1) (c) Four Reactor Coolant Pumps Operating	≥ 95% of minimum reactor coolant flow with 4 pumps operating *	≥ 95% of minimum reactor coolant flow with 4 pumps operating *	LA05 A06
Function 4	4. Pressurizer Pressure – High	≤ 2400 psia	≤ 2400 psia	
Function 5	5. Containment Pressure – High	≤ 3.3 psig	≤ 3.3 psig	
Function 6	6. Steam Generator Pressure – Low (2) (f)	≥ 600 psia	≥ 600 psia	
Functions 7a and 7b	7. Steam Generator Water Level – Low	≥ 35.0% Water Level – each steam generator	≥ 35.0% Water Level – each steam generator	
Function 8	8. Local Power Density – High (3) (g)(h)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.	L06

* For minimum reactor coolant flow with 4 pumps operating, refer to Technical Specification LCO 3.2.5. LA05

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

LA04

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
Function 9a	9. Thermal Margin/Low Pressure (1) (c) A06 Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 and ≥1887 psia M06
Function 9b	9a. Steam Generator Pressure Difference High (1) (c) (logic in TM/LP)	≤ 135 psid	≤ 135 psid
Function 10	10. Loss of Turbine – Hydraulic Fluid Pressure – Low (3) (g)(h)	≥ 800 psig	≥ 800 psig L06
Function 2	11. Rate of Change of Power – High (4) (d) Wide Range Logarithmic Neutron Flux Monitor Power	≤ 2.49 decades per minute	≤ 2.49 decades per minute

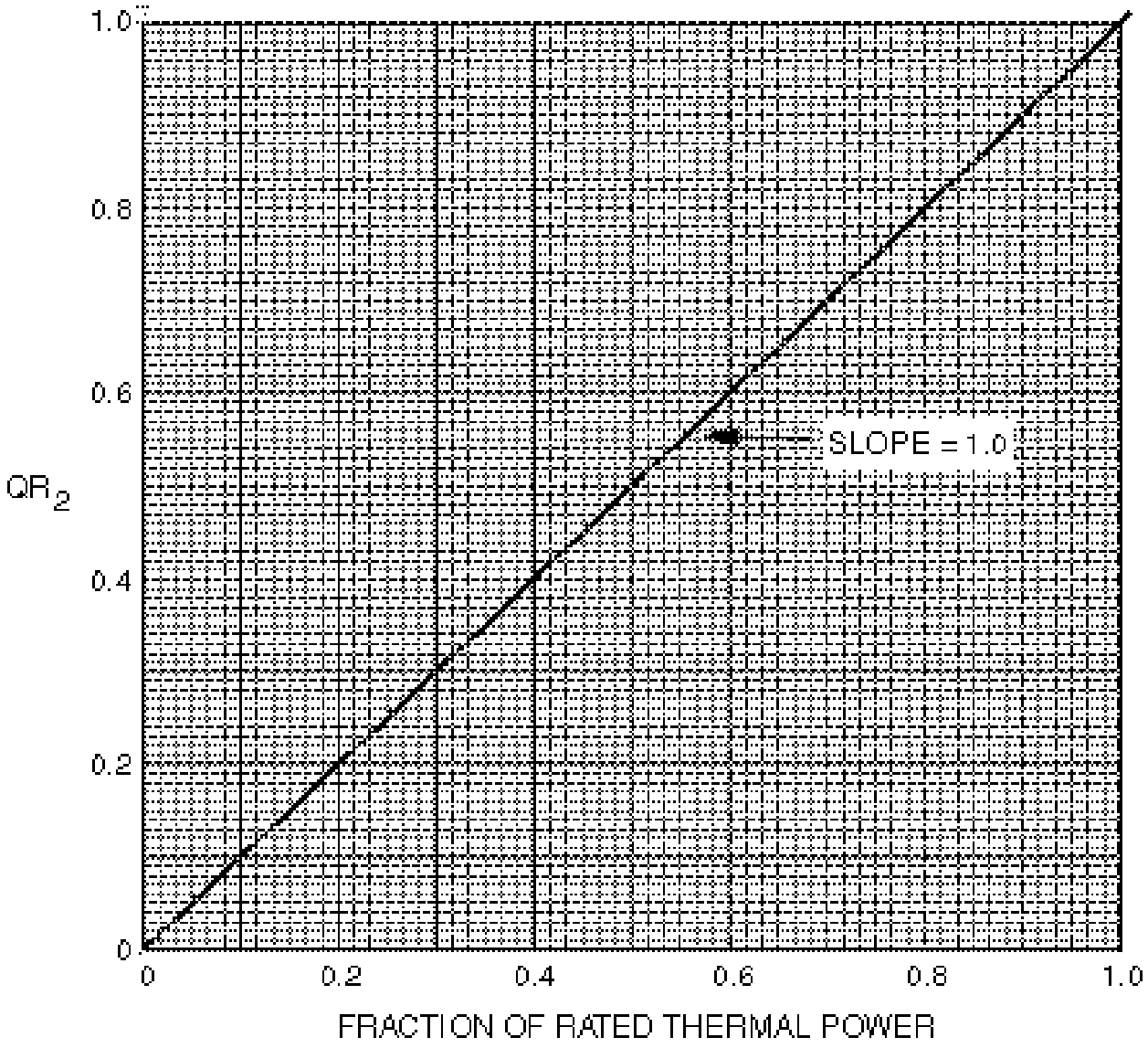
TABLE NOTATION

Footnote c (1) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is ≥ 1% of RATED THERMAL POWER.

Footnote f (2) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

Footnote g (3) Trip is not applicable and may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is ≥ 15% of RATED THERMAL POWER. L06

Footnote d (4) Trip may be bypassed below 10^{-4} % and above 15% of RATED THERMAL POWER.

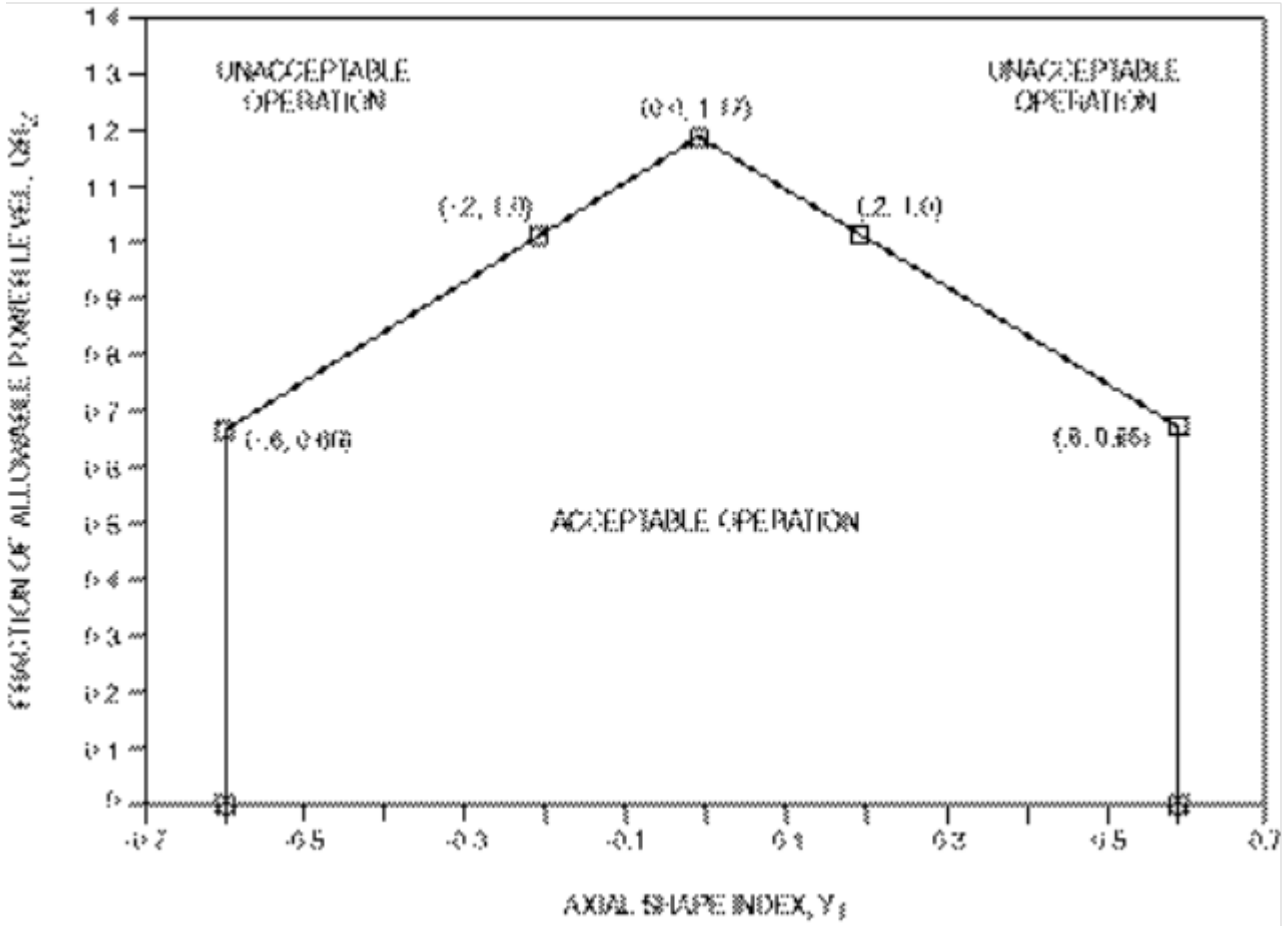


3.3.1-3 (Page 1 of 2)

FIGURE ~~2.2-1~~

Local Power Density – High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)

Figure 3.3.1-3

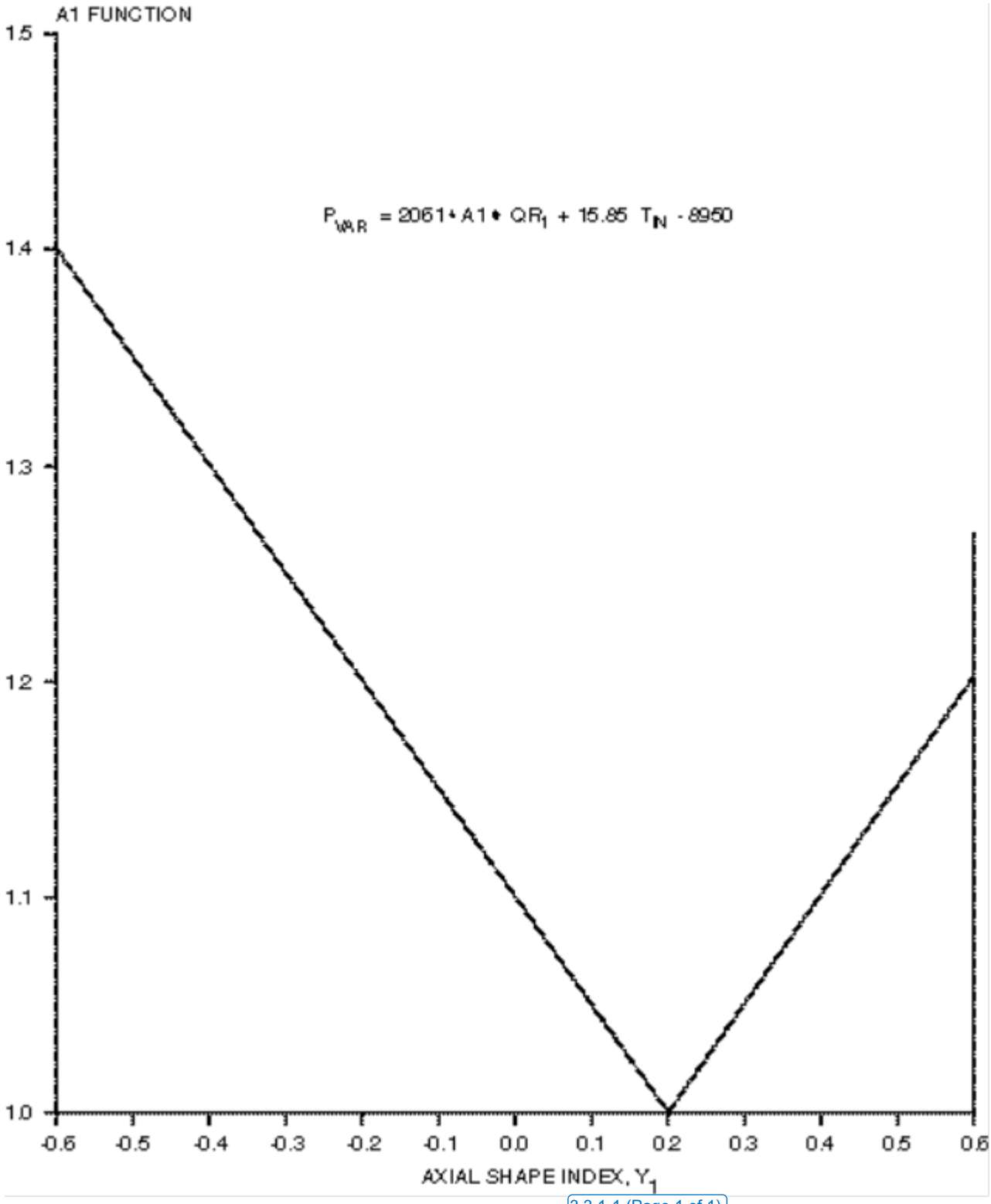


3.3.1-3 (Page 2 of 2)

FIGURE ~~2.2-2~~

Local Power Density – High Trip Setpoint Part 2 (QR_2 Versus Y_1)

Figure 3.3.1-1

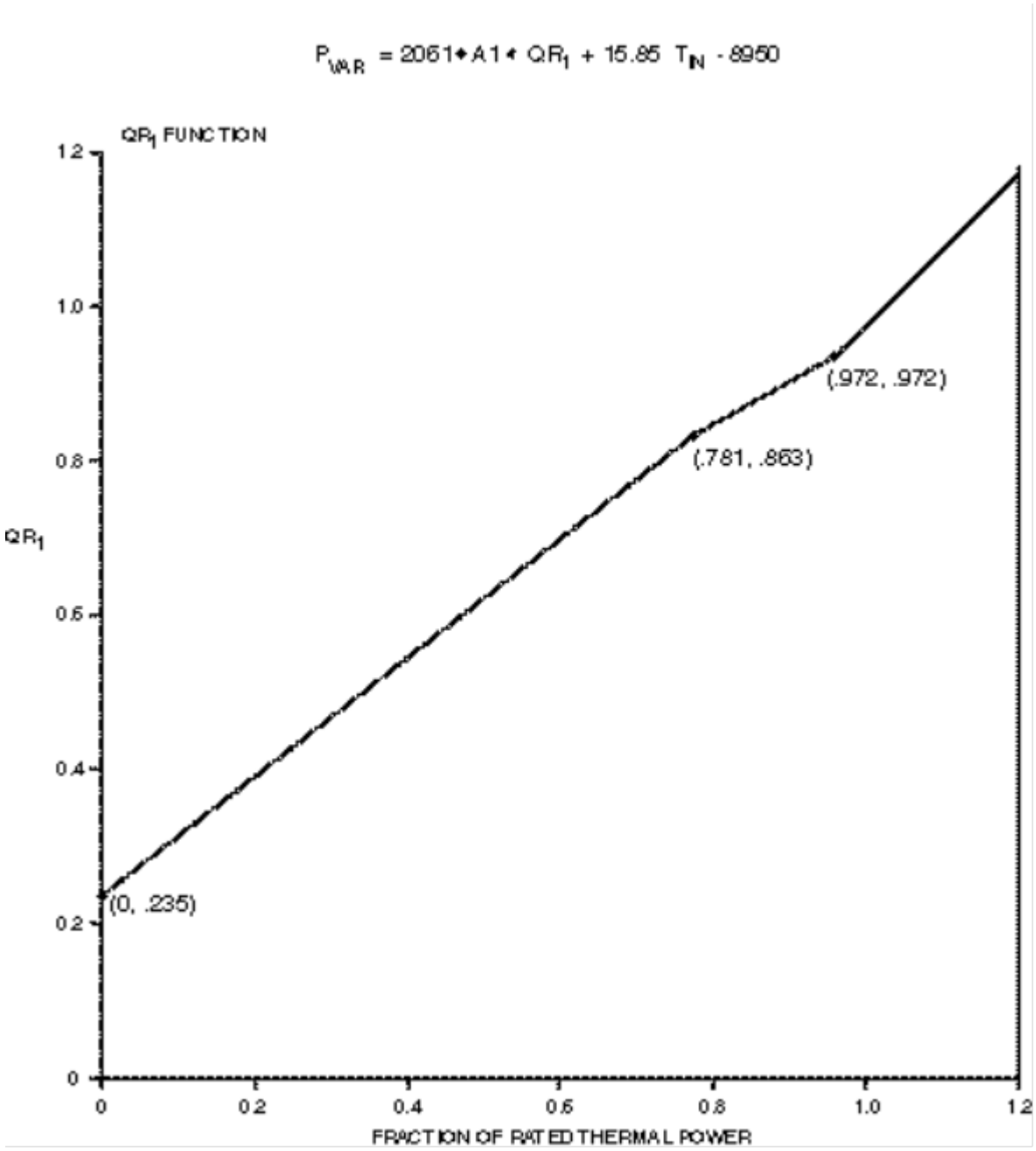


3.3.1-1 (Page 1 of 1)

FIGURE ~~2.2-3~~

Thermal Margin/Low Pressure Trip Setpoint

Figure 3.3.1-2



3.3.1-2 (Page 1 of 1)

FIGURE ~~2.2-4~~

Thermal Margin/Low Pressure Trip Setpoint
Part 2 (Fraction of RATED THERMAL POWER Versus QR_1)

3/4.3 INSTRUMENTATION

SYSTEM (RPS)

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.1

3.3.1 ~~As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE.~~

Four RPS trip units and associated instrument and bypass removal channels for each Function in Table 3.3.1-1

A02

Applicability **APPLICABILITY:** As shown in Table 3.3-1.

Add proposed ACTIONS note.

A03

ACTIONS **ACTION:**

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1
SR 3.3.1.2
SR 3.3.1.3
SR 3.3.1.4
SR 3.3.1.5
SR 3.3.1.6
SR NOTE

4.3.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

SR 3.3.1.7

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

SR 3.3.1.8

SR 3.3.1.9

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing. ~~Each test shall include at least one channel per function.~~

SR 3.3.1.9 NOTE

A04

ITS

A01

ITS 3.3.1

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

SYSTEM (RPS)

- OPERATING

OR OTHER SPECIFIED CONDITIONS

Table 3.3.1-1

LA01 LA01

A02

LCO 3.3.1 Applicability	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
	1. Manual Reactor Trip	4 4	2 2	4 4	1, 2 3*, 4*, 5*	1 5	See ITS 3.3.2
Function 1 Footnote c	2. Variable Power Level – High	4	(c) 2(a)(d) A09	3	1, 2	2	
Function 4	3. Pressurizer Pressure – High	4	2	3	1, 2	2	
Function 9a Footnote c	4. Thermal Margin/Low Pressure	4	(c) 2(a)(d) A09	3	1, 2	2	
Function 5	5. Containment Pressure – High	4	2	3	1, 2	2	
Function 6 Footnote f	6. Steam Generator Pressure – Low	4/SG	(f) 2/SG(b)	3/SG	1, 2	2	
Function 9b Footnote c	7. Steam Generator Pressure Difference – High	4	(c) 2(a)(d) A09	3	1, 2	2	
Function 7a and 7b	8. Steam Generator Level – Low	4/SG	2/SG	3/SG	1, 2	2	
Function 8 Footnote g and h	9. Local Power Density – High	4	(g) 2(e)(d) A09	3	1 (h)	2	L06
	10. ~~Loss of Component Cooling Water to Reactor Coolant Pumps~~	4	2	3	1, 2	2	R01
	11. Reactor Protection System Logic	4	2	3	1, 2 3*, 4*, 5*	2 5	See ITS 3.3.2
	12. Reactor Trip Breakers	4	2(f)	4	1, 2 3*, 4*, 5*	4 5	
Function 2 Footnotes d and e	13. Wide Range Logarithmic Neutron Flux Monitor a. Startup and Operating Power Rate of Change of Power – High	4	2(e)(g) LA02 (d)	3	1**, 2	2	
	b. Shutdown	4	0	2	3, 4, 5	3	See ITS 3.3.11
Function 3 Footnote c	14. Reactor Coolant Flow – Low	4/SG	2/SG(a)(d) (c) A09	3/SG	1, 2	2	
Function 10 Footnotes g and h	15. Loss of Load (Turbine Hydraulic Fluid Pressure – Low)	4	2(e) (g)	3	(h) 1	2	L06

Table 3.3.1-1

TABLE 3.3-1 (Continued)

TABLE NOTATION

See ITS 3.3.2

* With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

Footnote e

** ~~Mode 1 applicable only when Power Range Neutron Flux power ≤ 15% of RATED THERMAL POWER.~~

Trip not applicable when Thermal Power is > 15% RTP.

A01

Footnote c

(a) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER ~~in conjunction with (d) below~~; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of RATED THERMAL POWER.

A09

Footnote f

(b) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

L06

is not applicable and

Footnotes g & h

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is greater than or equal to 15% of RATED THERMAL POWER.

~~(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.~~

A09

Footnote d

(e) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq$ 10^{-4}% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

See ITS 3.3.2

~~(g) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.~~

LA02

h. Trip is only applicable in MODE 1 ≥ 15% RTP.

L06

ACTION STATEMENTS

See ITS 3.3.2

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

One or more Functions with one RPS trip unit or associated instrument channel inoperable except for Condition C (excore channel not calibrated with incore detectors).

----------NOTE----------
Required Action A.2 must be completed whenever this Condition is entered.

A10

ACTION 2 - a. ~~With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed~~ in ~~the~~ bypass~~ed~~ or trip~~ped condition~~ within 1 hour. ~~The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.~~

ACTION A Required Actions A.1 and A.2

Place affected trip unit

Restore channel to OPERABLE status prior to entering MODE 2 following next MODE 5 entry.

A01

ACTION B Required Action B.1

b. ~~With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:~~

One or more Functions with two RPS trip units or associated instrument channels inoperable except for Condition C (excore channel not calibrated with incore detectors).

A01

Place one trip unit in

trip unit

1. ~~Verify that one of the inoperable channels has been~~ bypass~~ed~~ and ~~place~~ the other ~~inoperable channel~~ in ~~the~~ trip~~ped condition~~ within 1 hour.

2. ~~All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition.~~

A07

~~With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:~~

~~Process Measurement Circuit~~	~~Functional Unit Bypassed~~
~~1. Safety Channel – Nuclear Instrumentation~~	
~~Wide Range~~	~~Rate of Change of Power – High (RPS)~~
~~Linear Range~~	~~Variable Power Level – High (RPS)~~ ~~Local Power Density – High (RPS)~~ ~~Thermal Margin/Low Pressure (RPS)~~
~~2. Pressurizer Pressure -~~	~~Pressurizer Pressure – High (RPS)~~ ~~Thermal Margin/Low Pressure (RPS)~~ ~~Pressurizer Pressure – Low (ESF)~~
~~3. Containment Pressure -~~	~~Containment Pressure – High (RPS)~~ ~~Containment Pressure – High (ESF)~~
~~4. Steam Generator Pressure -~~	~~Steam Generator Pressure – Low (RPS)~~ ~~Thermal Margin/Low Pressure (RPS)~~ ~~AFAS-1 and AFAS-2 (AFAS)~~ ~~Steam Generator Pressure – Low (ESF)~~
~~5. Steam Generator Level -~~	~~Steam Generator Level – Low (RPS)~~ ~~If SG-2A, then AFAS-1 (AFAS)~~ ~~If SG-2B, then AFAS-2 (AFAS)~~

LA06

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

~~ACTION 2 - (Continued)~~

~~6. Cold Leg Temperature~~	~~Variable Power Level – High (RPS)~~ ~~Thermal Margin/Low Pressure (RPS)~~ ~~Local Power Density – High (RPS)~~	LA06
~~7. Hot Leg Temperature~~	~~Variable Power Level – High (RPS)~~ ~~Thermal Margin /Low Pressure (RPS)~~ ~~Local Power Density – High (RPS)~~	

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes*. Verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

See ITS 3.3.11

See ITS 3.3.2

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour, provided the trip breakers of any inoperable channel are in the tripped condition, for surveillance testing per Specification 4.3.1.1.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

See ITS 3.3.11

* Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

Add proposed ACTION C — L04

Add proposed ACTIONS D and E. — L03

Add proposed ACTIONS F and G — A08

~~DELETED~~

~~DELETED~~

ITS

A01

ITS 3.3.1

Table 3.3.1-1

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

M01

LCO 3.3.1 Applicability

M02

FUNCTIONAL ~~UNIT~~	SR 3.3.1.1 CHANNEL CHECK	SR 3.3.1.2, SR 3.3.1.3, SR 3.3.1.5, SR 3.3.1.8 (a)(b) CHANNEL CALIBRATION	SR 3.3.1.4, SR 3.3.1.6, SR 3.3.1.7 CHANNEL FUNCTIONAL TEST (a)(b)	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
1. Manual Reactor Trip	N/A	N.A.	S/U(1)	1, 2, 3*, 4*, 5*	See ITS 3.3.2
Function 1 Footnote c 2. Variable Power Level – High					
a. Nuclear Power	SFCP	SFCP(2), SFCP(3), SFCP(4)	SFCP	1,2	
b. ΔT Power	SFCP	SFCP(5), SFCP(4)		1	
Function 4 3. Pressurizer Pressure – High	SFCP	SFCP	SFCP	1, 2	
Function 9a Footnote c 4. Thermal Margin/Low Pressure	SFCP	SFCP	SFCP	1, 2	
Function 5 5. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2	
Function 6 Footnote f 6. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2	
Function 9b Footnote c 7. Steam Generator Pressure Difference – High	SFCP	SFCP	SFCP	1, 2	
Functions 7a and b 8. Steam Generator Level – Low	SFCP	SFCP	SFCP~~(8, 9)~~	1, 2	
Function 8 Footnote g & h 9. Local Power Density – High	SFCP	SFCP	SFCP	1	
~~10. Loss of Component Cooling Water to Reactor Coolant Pumps~~	~~N.A.~~	~~N.A.~~	~~SFCP~~	~~N.A.~~	R01
11. Reactor Protection System Logic	N.A.	N.A.	SFCP(7)	1, 2, 3*, 4*, 5*	See ITS 3.3.2

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL ~~UNIT~~	SR 3.3.1.1 CHANNEL CHECK	SR 3.3.1.2, SR 3.3.1.3, SR 3.3.1.5, SR 3.3.1.8 CHANNEL CALIBRATION (a)(b)	SR 3.3.1.4, SR 3.3.1.6, SR 3.3.1.7 CHANNEL FUNCTIONAL TEST (a)(b)	MODES FOR WHICH SURVEILLANCE IS REQUIRED M01	M02
	12. Reactor Trip Breakers	N.A.	N.A.	S/U(1), SFCP, SFCP(6)	1, 2, 3*, 4*, 5*	See ITS 3.3.2
Function 2	13. Wide Range Logarithmic Neutron Flux Monitor ← Power Rate of Change - High	SFCP	SFCP	S/U(1),SFCP	1, 2, 3, 4, 5	See ITS 3.3.11 | R1
Function 3	14. Reactor Coolant Flow – Low	SFCP	SFCP	SFCP	1, 2	
Function 10	15. Loss of Load (Turbine Hydraulic Fluid Pressure – Low)	SFCP	SFCP → ~~N.A.~~ M03	~~SFCP~~	1	

Add proposed Note to SR 3.3.1.8 — L05

Once within 7 days prior to reactor startup — L07

R1

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - Only if the reactor trip breakers are in the closed position and the CEA drive system is capable of CEA withdrawal.

See ITS 3.3.2

SR 3.3.1.6 (1) - Each startup ~~or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal,~~ if not performed in the previous 7 days.

M04

Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP.

SR 3.3.1.2 NOTE 1 (2) - ~~Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Power – ΔT Power".~~ During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

SR 3.3.1.2 NOTE 2

M05

Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP.

SR 3.3.1.3 and NOTE ACTION C.2 (3) - ~~Above 15% of RATED THERMAL POWER,~~ recalibrate the excore detectors ~~which monitor the AXIAL SHAPE INDEX~~ by using the incore detectors or restrict THERMAL POWER ~~during subsequent operations~~ to ≤ 90% of the maximum allowed THERMAL POWER level ~~with the existing reactor coolant pump combination~~.

A05

SR 3.3.1.5 NOTE and SR 3.3.1.8 NOTE (4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

SR 3.3.1.2 (5) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

See ITS 3.3.2

(6) - In accordance with the Surveillance Frequency Control Program and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include verification of the independent OPERABILITY of the undervoltage and shunt trips.

(7) - The fuse circuitry in the matrix fault protection circuitry shall be determined to be OPERABLE by testing with the installed test circuitry.

instrument

Table 3.3.1-1 Footnote a (8) - If the as-found channel setpoint is ~~either~~ outside its predefined as-found acceptance criteria band ~~or is not conservative with respect to the Allowable Value~~, then the channel shall be ~~declared inoperable and shall be~~ evaluated to verify that it is functioning as required before returning the channel to service.

M01

Table 3.3.1-1 Footnote b (9) - ~~The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Field Trip Setpoint, otherwise that channel shall not be returned to OPERABLE status. The Field Trip Setpoint and the methodology used to determine the Field Trip Setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in UFSAR Section 7.2.~~

M02

The instrument channel setpoint shall be reset to a value that is within the as-left acceptance criteria band around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left acceptance criteria bands apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP and the methodologies used to determine the as-found and as-left acceptance criteria bands are specified in the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively.

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

LCO 3.3.1

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the ~~Trip Setpoint~~ values shown in Table 2.2-1.

Allowable

LA04

Applicability

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

ACTION A

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the ~~Trip Setpoint~~ value.

Allowable

LA04

Table 3.3.1-1

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

LA04

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
	1. Manual Reactor Trip	Not Applicable	Not Applicable
Function 1	2. Variable Power Level – High(1) (c)		
	Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER. A06
Function 4	3. Pressurizer Pressure – High	≤ 2370 psia	≤ 2374 psia
Function 9a	4. Thermal Margin/Low Pressure(1) (c)		
	Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia. A06
Function 5	5. Containment Pressure – High	≤ 3.0 psig	≤ 3.1 psig
Function 6	6. Steam Generator Pressure – Low	≥ 626.0 psia(2)	≥ 621.0 psia(2) (f)
Function 9b	7. Steam Generator Pressure Difference – High (Logic in TM/LP Trip Unit)(1) (c)	≤ 120.0 psid	≤ 132.0 psid
Functions 7a and 7b	8. Steam Generator Level – Low	≥ 35.0%(3)	≥ 35.0%(3) LA07

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

LA04

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
Function 8 — 9. Local Power Density – High(5) (g)(h) Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2. L06
10. Loss of Component Cooling Water to Reactor Coolant Pumps – Low	≥ 636 gpm**	≥ 636 gpm R01
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
Function 2 — Wide Range Logarithmic Neutron Flux Monitor Power — 13. Rate of Change of Power – High(4) (d)	≤ 2.49 decades per minute	≤ 2.49 decades per minute
Function 3 — 14. Reactor Coolant Flow – Low(1) (c)	≥ 95.4% of minimum Reactor Coolant flow with four pumps operating*	≥ 94.9% of minimum Reactor Coolant flow with four pumps operating* LA05
Function 10 — 15. Loss of Load (Turbine) Hydraulic Fluid Pressure – Low(5) (g)(h)	≥ 800 psig	≥ 800 psig L06

* For minimum reactor coolant flow with four pumps operating, refer to Technical Specification LCO 3.2.5. LA05

** 10-minute time delay after relay actuation. LA04

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

A09

Footnote c (1) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER ~~during testing pursuant to Special Test Exception 3.10.3~~; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of ~~-~~RATED THERMAL POWER.

Footnote f (2) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

~~(3) % of the narrow range steam generator level indication.~~

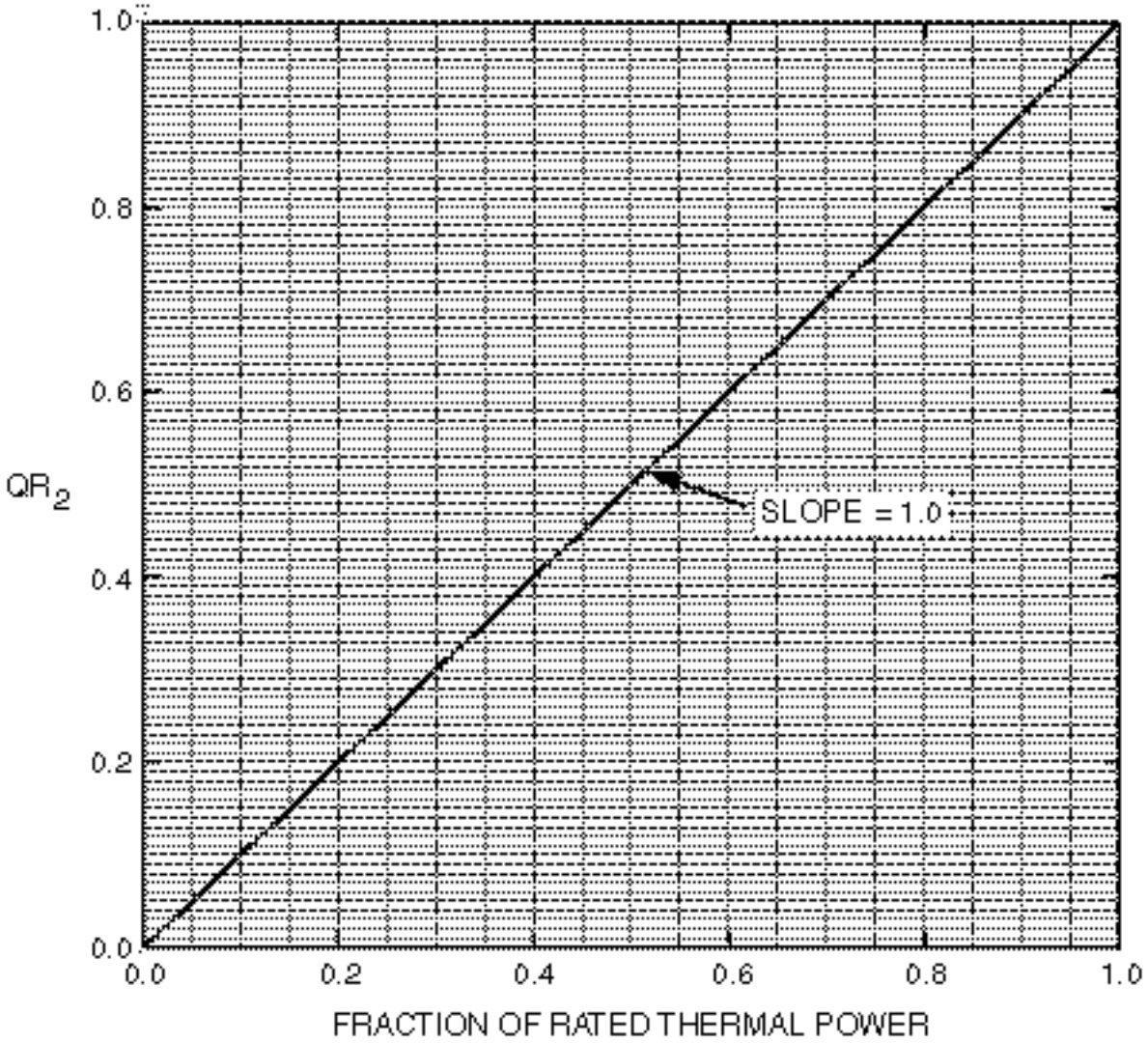
LA07

Footnote d (4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed ~~-~~when Wide Range Logarithmic Neutron Flux power is $\geq$ 10^{-4}% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

L06

is not applicable and

Footnotes g and h (5) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron~~-~~ Flux power is greater than or equal to 15% of RATED THERMAL POWER.



3.3.1-3 (Page 1 of 2)

FIGURE ~~2.2-1~~

Local power density – High trip setpoint
Part 1 (Fraction of RATED THERMAL POWER versus QR_2)

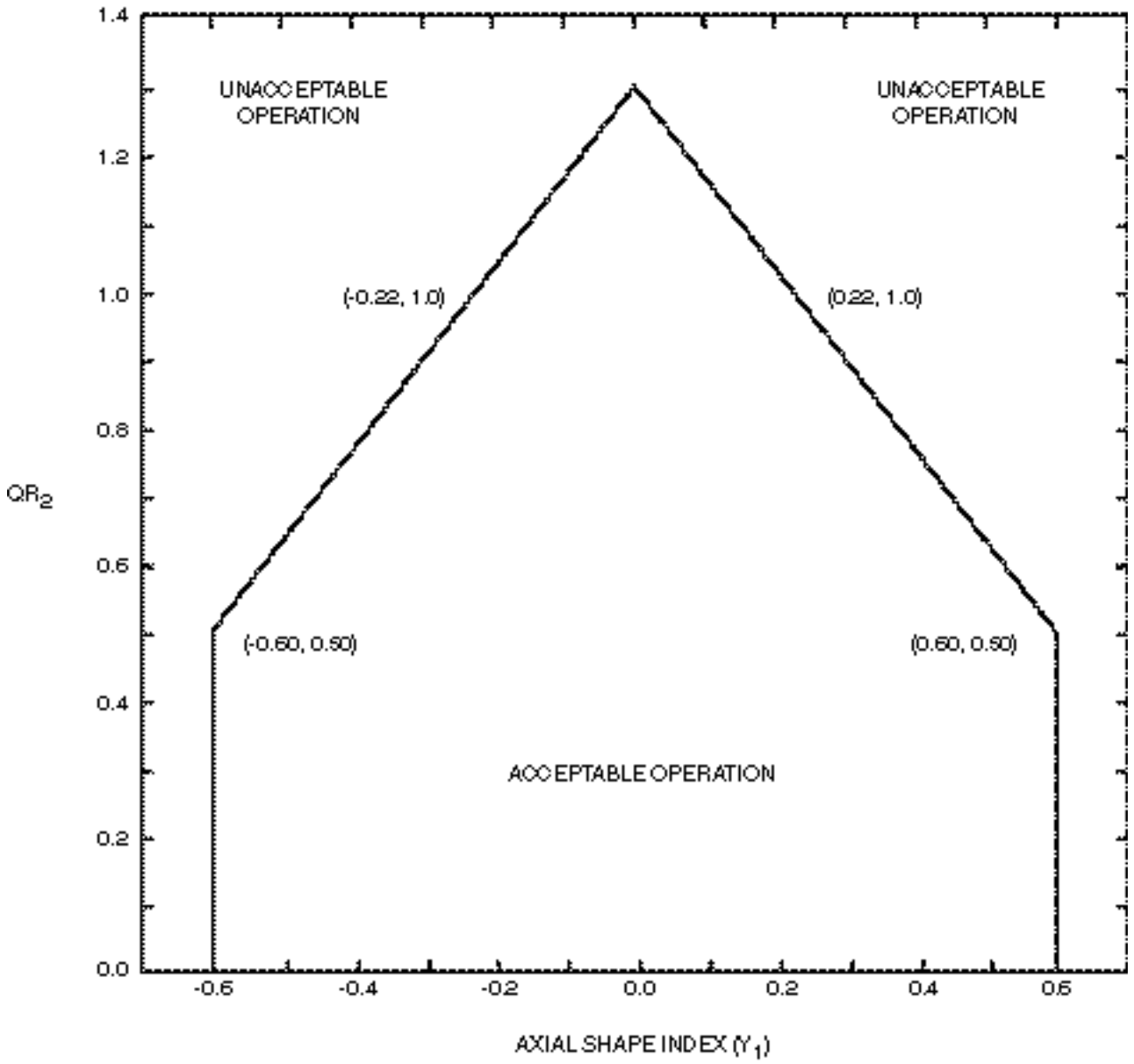


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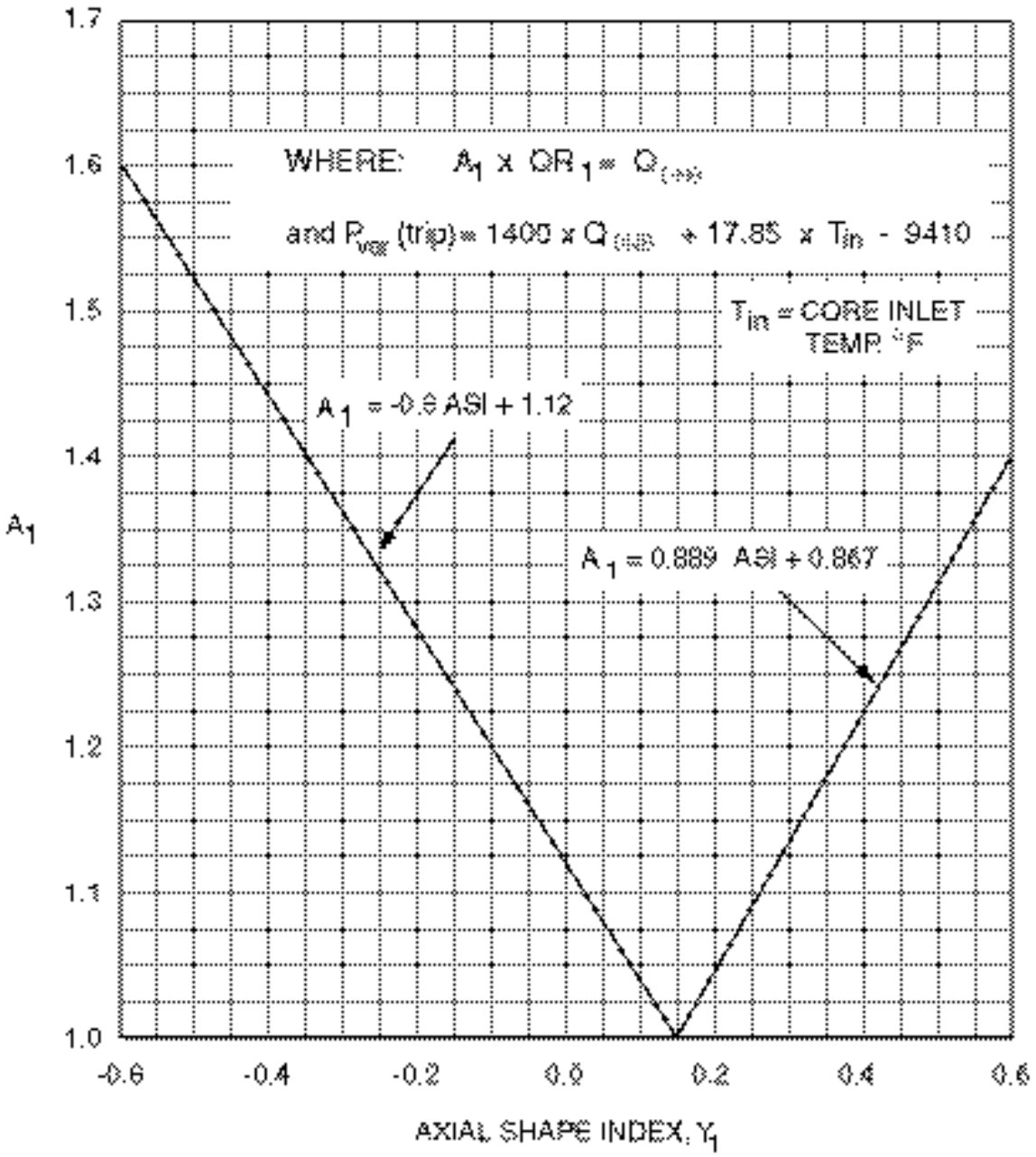
FIGURE ~~2.2-2~~

LOCAL POWER DENSITY – HIGH TRIP SETPOINT PART 2
(QR_2 versus Y_1)



3.3.1-1 (Page 1 of 1)

FIGURE ~~2.2-3~~

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 1 (Y_1 Versus A_1)

WHERE: $A_1 \times QR_1 = Q_{DNB}$

AND $p_{var}^{trip} = 1400 \times Q_{DNB} + 17.85 \times T_{in} - 9410$

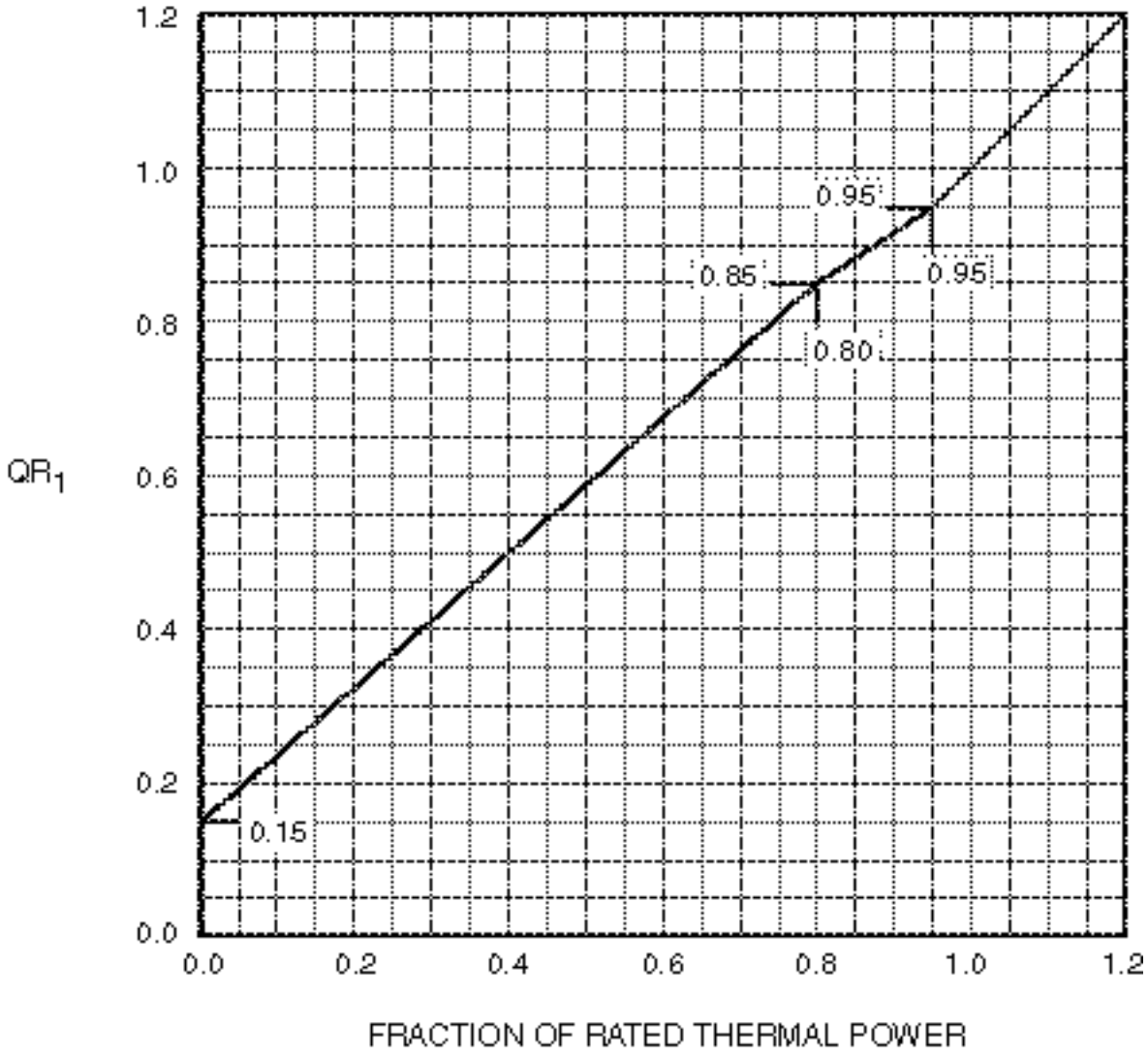


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FIGURE ~~2.2-4~~

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 2 (FRACTION OF RATED THERMAL POWER VERSUS QR_1)

DISCUSSION OF CHANGES
ITS 3.3.1, REACTOR PROTECTIVE SYSTEM (RPS) INSTRUMENTATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS Table 3.3-1 specifies the "TOTAL NO. OF CHANNELS" and the "MINIMUM CHANNELS OPERABLE" associated with each RPS Functional Unit. For Unit 1 CTS Table 3.3-1 Functional Units 2, 3, 4, 7, 8, 9, 9a, and 11a and for Unit 2 Functional Units 2 through 10, 13a, 14 and 15; the number of channels listed in the "TOTAL NO. OF CHANNELS" column is greater than that listed in the "MINIMUM OPERABLE CHANNELS" column. CTS Table 3.3-1 Action 2 specifies the actions to take with the number of channels OPERABLE one less than required by the "TOTAL NO. OF CHANNELS" column. ITS LCO 3.3.1 requires “Four RPS trip units and associated instrument and bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.” The ITS 3.3.1 ACTIONS require entry when the OPERABLE channels are one less than required by the LCO statement. This changes the CTS by deleting the "MINIMUM CHANNELS OPERABLE" column and utilizing the channel requirements specified in the TOTAL NO. OF CHANNELS column in ITS LCO 3.3.1.

This change is acceptable because the requirements for when actions must be taken remain unchanged. The ITS LCO statement reflects the current requirements in the CTS Actions for when actions are required to be taken. The "MINIMUM CHANNELS OPERABLE" column for CTS Table 3.3-1 has been removed and the "TOTAL NO. OF CHANNELS" column numbers are reflected in ITS LCO 3.3.1. This change is a presentation preference consistent with NUREG-1432, Rev. 5 and PSL CTS Table 3.3-1 Functional Unit Total No. of Channels to Trip column. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 Unit 1 CTS 3.3.1.1 ACTION, Unit 2 CTS 3.3.1 ACTION and CTS Table 3.3-1 provide the compensatory actions to take when RPS instrumentation is inoperable. ITS 3.3.1 ACTIONS provide the compensatory actions for inoperable RPS Instrumentation. The ITS 3.3.1 ACTIONS includes a Note that allows separate Condition entry for each Function. This modifies the CTS by providing a specific allowance to enter the Action for each inoperable RPS instrumentation Function and for certain Functions on a steam generator basis.

The purpose of the CTS Actions is to provide the appropriate compensatory actions for inoperable RPS Functions. This proposed change will allow separate condition entry for each RPS instrumentation Function. This change is acceptable because it clearly states the current requirement. The CTS considers each RPS instrumentation Function to be separate and independent from the

others. In addition, the channels associated with Functions 3 and 6 are allowed separate Condition entry on the specified basis (i.e., per SG) since the channels associated with each steam generator will provide the associated RPS trip based on the logic associated with the channels on the specified basis. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 Unit 1 CTS 4.3.1.1.3 and Unit 2 CTS 4.3.1.3 state, in part, that the RPS RESPONSE TIME test of each reactor trip function "shall include at least one channel per function." ITS SR 3.3.1.9 requires performance of an RPS RESPONSE TIME test but does not explicitly state that the RPS RESPONSE TIME test must include at least one channel per function.

The purpose of the CTS statement in the Surveillance Requirement associated with the RPS RESPONSE TIME test is to ensure required features are tested to verify the RPS instrument Functions are OPERABLE. However, this statement is redundant to the definition of RPS RESPONSE TIME provided in ITS Section 1.1 and therefore is not necessary.

The definition of the RPS RESPONSE TIME states, in part, that the test, "…shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. This encompasses the requirement that at least one channel per Function be included in the RPS RESPONSE TIME. The required testing will continue, per the ITS SR wording and the definition provided in Section 1.1, to include the features necessary to verify the RPS instrument Functions are OPERABLE.

This change is designated as administrative because it does not result in technical changes to the CTS.

A05 CTS Table 4.3-1 footnote 3 states, in part, "with the existing Reactor Coolant Pump combination." ITS SR 3.3.1.3 and Required Action C.2 do not include this specific detail. This changes CTS by removing redundant detail. This change is acceptable because reactor coolant pump (RCP) OPERABILITY and required operation is addressed in other Specifications in CTS and retained in ITS 3.4.1 and 3.4.4. LCO 3.4.1 and LCO 3.4.4 require, in part, a minimum reactor coolant flow and both Reactor Coolant System (RCS) loops to be in operation, respectively (i.e., all four RCPs are in operation) in MODES 1 and 2. The "existing Reactor Coolant Pump combination" is determined by compliance with these LCOs and retention of the information in ITS SR 3.3.1.3 is unnecessary.

This change is acceptable because the requirements for RCP operation remain unchanged and are retained in ITS 3.4.1 and 3.4.4. This change is designated as administrative because it does not result in a technical change to CTS.

A06 CTS Table 2.2-1 Functional Units 2, 3 and 9 (Unit 1) and Functional Units 2 and 4 (Unit 2) specify "Four Reactor Coolant Pumps Operating". The equivalent RPS Functions in ITS Table 3.3.1-1 Functions 1, 3, and 9a do not include this detail. Requirements for reactor coolant pump OPERABILITY and required operation are provided in CTS 3.4.1.1. LCO 3.4.1 and LCO 3.4.4 require, in part, a

minimum reactor coolant flow and both Reactor Coolant System (RCS) loops to be in operation, respectively (i.e., all four RCPs are in operation) in MODES 1 and 2. This changes the CTS by removing the "Four Reactor Coolant Pumps Operating" redundant detail from the RPS specification.

This change is acceptable because the requirements for reactor coolant pump operation remain unchanged. This change is designated as administrative because it does not result in a technical change to CTS.

A07 Unit 1 CTS Table 3.3-1, Action 2.b and Unit 2 CTS Table 3.3-1, Action 2.b.2 state, in part, that all functional units receiving an input from (or affected by) the inoperable channel are (shall be) also bypassed or tripped. ITS 3.3.1 ACTIONS for each Function include a Required Action to place the affected trip unit in trip or bypass within 1 hour. CTS LCO 3.0.2 and ITS LCO 3.0.2 establish that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. Therefore, when an instrument channel becomes inoperable, LCO 3.0.2 requires each Function affected by the inoperability shall be declared inoperable and the associated ACTIONS shall be met. Retention of these specific CTS Actions are not necessary and are deleted. This change is designated as administrative because it does not result in a technical change to the CTS.

A08 Unit 1 CTS 3.3.1.1 and Unit 2 CTS 3.3.1 do not contain a default action to perform when the Table 3.3-1 Actions cannot be completed within the required time; therefore, an entry into CTS 3.0.3 would be required. ITS 3.3.1 ACTION F requires reducing THERMAL POWER to < 15% RTP in 6 hours when Required Action and associated Completion Time for Loss of Power Density and Loss of Load Trip Functions are not met. ITS 3.3.1 ACTION G requires the unit to be in MODE 3 in 6 hours when Required Action and associated Completion Time for RPS Functions other than the Loss of Power Density and Loss of Load Trip Functions are not met. (See DOC L06 for discussion of modification to Applicability) This changes the CTS by explicitly stating the required ACTIONS when actions cannot be performed within the required time. R1

The purpose of ITS 3.3.1 ACTIONS F and G is to ensure the plant is brought to a MODE in which the LCO for the RPS Functions does not apply within a reasonable amount of time in a controlled manner because the Technical Specification actions cannot be completed as required. Unit 1 CTS 3.3.1.1 and Unit 2 CTS 3.3.1 are silent on these actions, deferring to CTS 3.0.3 for the actions to accomplish this. With the exception of the Loss of Power Density and Loss of Load Trip Functions, the RPS Functions are required to be OPERABLE in MODES 1 and 2. The Loss of Power Density and Loss of Load Trip Functions are required to be OPERABLE with THERMAL POWER ≥ 15% RTP per ITS Table 3.3.1-1 Footnote (h) (CTS Table 3.3-1 Note (c)). The proposed change is acceptable because the ACTIONS specified in ITS 3.3.1 adopt the ISTS structure for placing the unit outside the MODE of Applicability without changing the time to be below 15% RTP for the Loss of Power Density and Loss of Load Trip Functions and the time specified to enter MODE 3 for the other RPS Functions. Default Conditions eliminate the need to enter LCO 3.0.3 when one or more Required Actions cannot be met within the associated Completion

Times. This change is designated as administrative because it does not result in a technical change to the CTS.

A09 **Unit 2 only:** Current Technical Specifications (CTS) 3/4.10.3 provides an exception to the limitations of CTS 3.4.1 and the requirements of CTS Tables 2.2-1 and 3.3-1. This special test exception permits reactor criticality under reduced flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels. ITS does not contain this special test exception. This changes the CTS by eliminating a special test exception discussion from the footnotes.

This change is acceptable because this type of PHYSICS TEST is no longer performed. As a result, this CTS Special test exception is not needed and the references to the test are removed. This change is designated as administrative because it does not result in a technical change to the CTS and is consistent with the changes addressed in CTS 3/4.10.3.

A10 **Unit 2 only:** CTS 3.3.1 Table 3.3-1 Action 2a provides actions when one channel associated with the RPS Functional Units (except 1, 11, 12, and 13.b) is inoperable. The CTS Action states in part, “The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.” ITS 3.3.1 ACTION A includes this action to restore the channel to OPERABLE with a Completion Time of, “Prior to entering MODE 2 following the next MODE 5 entry.” This presentation preference is similar to the Completion Time specified in ISTS 3.3.1 (Digital), Required Action A.2. ITS 3.3.1, Condition A is modified by a Note which requires the action to restore the channel to OPERABLE status within the Completion Time to be performed whenever the Condition is entered. This changes the CTS by explicitly stating that the Required Action must be performed within the Completion Time whenever the condition is entered.

This change is necessary to eliminate the conflict with the general CTS 3.0.2 (ITS LCO 3.0.2) requirement that if the LCO is met or is no longer applicable prior to expiration of the specified time interval(s), completion of the ACTIONS is not required, unless otherwise stated. The addition of the Note constitutes an “unless otherwise stated.” The Note will require the action to be completed within the Completion Time even when the LCO is no longer applicable, to ensure the channel is restored to OPERABLE prior to the next reactor startup from MODE 5 conditions. This change is acceptable because it reflects the current understanding and application of the CTS Action requiring the applicable channel to be restored to OPERABLE status whenever the LCO is not met.

This change is designated as administrative as it clarifies the current understanding of the CTS requirement while providing an “unless otherwise stated,” to the requirements of CTS 3.0.2 (ITS LCO 3.0.2).

MORE RESTRICTIVE CHANGES

M01 CTS 2.2.1 ACTION states, in part, that with a RPS instrumentation or interlock setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable. ITS Table 3.3.1-1 Footnote (a) states, "If the as-found channel setpoint is outside its predefined as-found

acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service." ITS Bases Section 3.3.1, referring to the Allowable Values, states: "This value is specified in the Technical Specifications in order to define OPERABILITY of the channels and is designated as the Allowable Value." ITS Section 3.3.1 Bases also states, "If the actual setting (as-found setpoint) of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found acceptance criteria band (e.g., conservative side of the as-found acceptance criteria band), the channel is OPERABLE, but degraded. The degraded condition will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the NTSP (within the allowed acceptance criteria band) and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance." CTS Table 4.3-1 Note (8) is similar to the proposed footnote but applies to only the Steam Generator Level – Low RPS Function. This changes the CTS by requiring an evaluation of channel functionality (extent of which is expanded on in the TS Bases) prior to returning it to service in addition to the as-found value being conservative to the Allowable Value and applies to all the RPS Function channels.

The purpose of this Note is to address a concern that the Technical Specification requirements for Limiting Safety System Settings (LSSS) may not be fully in compliance with the intent of 10 CFR 50.36. Specifically, the concern is that the existing Surveillance Requirements may not provide adequate assurance that instruments will actuate safety functions at the point assumed in the applicable safety analysis. 10 CFR 50.36(c)(1)(ii)(A) states, "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the settings must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor." The proposed change clarifies the Technical Specification requirements to ensure that the automatic protective action will correct the abnormal situation before a safety limit is exceeded. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. This change is consistent with the ISTS. This change is considered a more restrictive change because additional requirements have been added to Surveillance Requirements.

M02 CTS 2.2.1 ACTION states, in part, to declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 (Unit 1) and 3.3.1 (Unit 2) until the channel is restored to OPERABLE status "with its trip

setpoint adjusted consistent with the Trip Setpoint value." ITS Table 3.3.1-1 Footnote (b) states, "The instrument channel setpoint shall be reset to a value that is within the as-left acceptance criteria band around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left acceptance criteria bands apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP and the methodologies used to determine the as-found and as-left acceptance criteria bands are specified in the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively." CTS Table 4.3-1 Note (9) is similar to the proposed footnote but applies to only the Steam Generator Level – Low RPS Function. This changes the CTS by providing more detailed information describing what "consistent with the Trip Setpoint" means and states a specific location where the NTSPs and the methodology for determining the double-sided NTSP instrument tolerance bands are located. This additional footnote applies to all the RPS Function channels.

The purpose of this Note is to address a concern that the Technical Specification requirements for LSSS may not be fully in compliance with the intent of 10 CFR 50.36. Specifically, the concern is that the existing Surveillance Requirements may not provide adequate assurance that instruments will always actuate safety functions at the point assumed in the applicable safety analysis. 10 CFR 50.36(c)(1)(ii)(A) states, "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the settings must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor." The proposed change clarifies the Technical Specification requirements to ensure that the automatic protective action will correct the abnormal situation before a safety limit is exceeded. This change is consistent with the ISTS. This change is considered a more restrictive change because additional requirements have been added to Surveillance Requirements.

M03 CTS Table 4.3-1 Functional Unit 10 (Unit 1) and Functional Unit 15 (Unit 2) Loss of Load channels and Unit 1 Functional Unit 11 Wide Range Logarithmic Neutron Flux Monitor channels do not require Channel Calibration. ITS SR 3.3.1.8 requires Channel Calibration of each appliable RPS instrument channel including bypass removal functions with a periodic surveillance frequency in accordance with the Surveillance Frequency Control Program (SFCP). A note to the SR excludes neutron flux detectors from the Channel Calibration. This changes the CTS by adding an additional surveillance requirement for the Wide Range Logarithmic Neutron Flux Monitor Power Rate of Change – High (ITS Table 3.3.1-1 Function 2) and the Loss of Load (ITS Table 3.3.1-1 Function 10 (Unit 1) and Function 15 (Unit 2)) RPS Functions.

The proposed change is acceptable because it provides additional assurance that the Wide Range Logarithmic Neutron Flux Monitor Power Rate of Change -

High and Loss of Load RPS Functions are functioning properly. This change is consistent with ISTS.

PSL controls periodic Frequencies for Surveillances in accordance with the SFCP per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, the initial Frequency in accordance with the Surveillance Frequency Control Program will be specified as 18 months consistent with ISTS 3.3.1.8. The periodic Frequency is acceptable based upon the assumption of a 22.5 month calibration interval (18 months plus 25% grace) for the determination of the magnitude of equipment drift and is consistent with the CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST Frequencies for the PSL Unit 2 wide range neutron monitoring instrumentation.

The SFCP was established as described in FPL (PSL Unit 1 and Unit 2) "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program" (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066).

The proposed change is designated more restrictive because it includes additional requirements that are not specified in the CTS.

M04 The CTS channel calibration surveillance assigned to the power range neutron flux high setpoint on Table 4.3-1 consists of adjusting the power range instrument channels to match the calorimetric heat balance calculation. The CTS surveillance is modified by Note (2) that states the surveillance is only required above 15% RTP without specifying a time for first calibration during power escalation. ITS SR 3.3.1.2 is modified by a Note that states the surveillance is not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. The CTS is revised to be consistent with the ISTS. This changes the CTS by adding more specific requirements for the performance of the adjustment of the power range channels to the results of the calorimetric heat balance calculation.

The proposed change provides additional guidance for performing the power range instrumentation surveillance. The ISTS note establishes 12 hours for performing the first adjustment of the power range channels during power escalation. The proposed change is acceptable because it provides additional assurance that the power range instrument adjustments are accomplished in a timely and controlled manner such that the accuracy of the power range instrumentation is maintained within acceptable limits relative to the applicable safety analysis assumptions. As such, the proposed change enhances the assurance provided by the TS that the affected plant equipment is maintained OPERABLE and that the plant continues to be operated in a safe manner. The proposed change is designated more restrictive because it includes a specific additional requirement that is not specified in the CTS.

M05 The CTS channel calibration requirement associated with the power range neutron flux high setpoint on Table 4.3-1 is modified by Note (3). CTS Table 4.3-1 Note (3) specifies that the surveillance recalibrate the excore detectors using the incore detectors above 15% RTP without specifying a first performance time. ITS SR 3.3.1.3 is modified by a Note that states the surveillance is not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. The CTS is revised to conform to the proposed ISTS surveillance requirement. This changes the CTS by placing a new time restriction on the first performance of the surveillance after exceeding the specified power.

The first performance of this surveillance after a shutdown or refueling outage is important to establish the instrument accuracy is within acceptable tolerances. The proposed change is acceptable because it provides assurance that the surveillance is completed in a timely manner and that the instrument accuracy is verified within acceptable limits shortly after a plant startup. As such, the proposed change enhances the assurance provided by the TS that the instrument is maintained operable and that the plant continues to be operated in a safe manner. The proposed change is designated more restrictive because it includes additional requirements that are not specified in the CTS.

M06 **Unit 1 only:** CTS Table 2.2-1 Functional Unit 9 does not include a minimum allowable value for the reactor coolant pressure (RCS). The thermal margin/low pressure trip allowable value is determined based on a RCS minimum pressure or a computed value that is a function of the higher ΔT power or neutron power, reactor inlet temperature, the number of coolant pumps operating and the AXIAL SHAPE INDEX. The proposed ITS 3.3.1 Function 9a adds the minimum pressure value of 1887 psia to the allowable values determined by ITS Figures 3.3.1-1 and 3.3.1-2. This changes the CTS Table 2.2-1 Functional Unit 9 allowable value by adding a minimum pressure value.

The purpose of the thermal margin/low pressure trip is to protect against the occurrence of DNB during steady state operation. This change is acceptable because the RCS pressure (pressurizer pressure) minimum value is an assumption considered in the accident analysis. In addition, the addition of the allowable value is consistent with the PSL Unit 2 thermal margin/low pressure trip allowable value. This changes CTS by adding an allowable value requirement not currently required in CTS. The proposed change is designated more restrictive because it includes additional requirements that are not specified in the CTS.

RELOCATED SPECIFICATIONS

R01 **Unit 2 only:** CTS Tables 2.2-1, 3.3-1, and 4.3-1 provides requirements, in part, for the Loss of Component Cooling Water to Reactor Coolant Pumps – Low reactor trip function. The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary (RCPB) during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents. The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by

specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as Limiting Conditions for Operation on other reactor system parameters and equipment performance. Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions.

The Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function specified in CTS Table 2.2-1, 3.3-1 and 4.3-1 is provided as a plant specific reactor trip as an equipment protective feature required as a stipulation by NRC staff in justifying acceptance of the Reactor Coolant Water Component Cooling Water piping design. This reactor trip is not required for reactor protection and is not assumed in any accident or transient analysis. In addition, this automatic reactor trip feature on a loss of component cooling water (CCW) flow to the reactor coolant pumps (RCPs) is not required to protect against violating the core specified acceptable fuel design limits or breaching the RCPB during AOOs and is not considered an LSSS for a variable that has a significant safety function. As a result, the Loss of Component Cooling Water to Reactor Coolant Pumps – Low reactor trip function will be relocated from the Technical Specifications to the Technical Requirements Manual (TRM).

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary.

2. The Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

3. The Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function is not a structure, system, or component that is part of the primary success path or which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The loss of CCW flow to the RCPs is addressed in the PSL PRA with respect to requiring operator action to recover the CCW flow or remove the RCPs from service. The Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function is not a significant risk contributor to core damage frequency and offsite releases in the PSL PRA. Therefore, the Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function does not represent a structure, system, or component which operating experience

or probabilistic risk assessment has shown to be significant to public health and safety.

Since the selection criteria have not been satisfied for Loss of Component Cooling Water to Reactor Coolant Pumps – Low automatic reactor trip function, the requirements of this reactor trip instrument may be relocated to a licensee controlled document outside the Technical Specifications. Other reactor trip functions that are necessary to protect against violating the core specified acceptable fuel design limits or breaching the RCPB during AOOs are retained in the Technical Specifications (e.g., Reactor Coolant Flow – Low reactor trip and manual reactor trip functions). The Technical Specification requirements for the RPS Loss of Component Cooling Water to Reactor Coolant Pumps – Low Function will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as a relocation because the requirements for this instrument do not meet the criteria in 10 CFR 50.36(c)(2)(ii) and have been relocated to the TRM.

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-1 for RPS instrumentation has three columns stating various requirements for each function. These columns are labeled, "TOTAL NO. OF CHANNELS," "CHANNELS TO TRIP," and "MINIMUM CHANNELS OPERABLE." ITS Table 3.3.1-1 does not retain the "TOTAL NO. OF CHANNELS" or "CHANNELS TO TRIP" columns. This changes the CTS by moving the information of the "TOTAL NO. OF CHANNELS" and "CHANNELS TO TRIP" columns to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the number of required channels and the appropriate Condition to enter if a required channel becomes inoperable. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.3.1 Table 3.3-1 Unit 1 footnote (f) and Unit 2 footnote (g) for Unit 1 CTS Function 2 and Unit 2 CTS Function 13a, respectively, state “There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.” ITS 3.3.1 Functions 1 and 2 do not include this information. The ITS only specifies that channels be OPERABLE. This changes the CTS by moving the design/operational detail to the UFSAR.

The removal of these details, related to system design, from Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The excore instrumentation overlap requirement is a system design detail not related to any RPS reactor trip function. ITS still retains the requirements for Wide Range and Power Range monitoring channels to be OPERABLE. In addition, overlap of the range is checked during power transition to ensure the design requirement for continuous monitoring over the full reactor power range. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive change because information relating to system design is being removed from the Technical Specifications.

LA03 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 4.3-1 Functional Unit 2 Note (3) states, in part, "recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX." ITS SR 3.3.1.3 does not contain the system description detail "which monitor the AXIAL SHAPE INDEX." This changes the CTS by moving the system description detail that the excore detectors monitor the AXIAL SHAPE INDEX to the UFSAR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirements for performance of the surveillance requirement. Also, this change is acceptable because the removed information is adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA04 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements*) CTS 2.2.1 requires the RPS instrumentation setpoints to be set consistent with the Trip Setpoint values shown in Table 2.2-1. However, the CTS 2.2.1 Action is only required to be taken when the setpoint is less conservative than the Allowable Value column of Table 2.2-1. When the setpoint is less conservative than the Allowable Value, the channel is to be declared inoperable and adjusted consistent with the Trip Setpoint value. CTS Table 2.2-1 specifies both the Trip Setpoints and Allowable Values for the RPS Instrumentation Functional Units. ITS 3.3.1 requires the RPS instrumentation for each Function in Table 3.3.1-1 to be OPERABLE. ITS Table 3.3.1-1 specifies only the Allowable Values for the RPS Instrumentation Functions. The Allowable Values represent the OPERABILITY limit of the channels in ITS. This changes the CTS by moving the Trip Setpoints to the Technical Requirements Manual (TRM).

The purpose of the trip setpoint requirements is to ensure required automatic safety systems are actuated to protect against violating core design limits, breaching the Reactor Coolant System pressure boundary, and to mitigate accidents. Pursuant to 10 CFR 50.36(c)(1)(ii)(A), if it is determined that an

automatic protective device for a variable on which a safety limit has been placed (i.e., limiting safety system setting) does not function as required, appropriate action is taken to ensure the abnormal situation is corrected before a safety limit is exceeded, which may include shutting down the reactor. The PSL Instrument Setpoint Methodology calculates nominal trip setpoints (NTSPs) using methods consistent with the guidance provided in NRC Regulatory Guide (RG) 1.105, "Setpoints For Safety-Related Instrumentation," and ANSI/ISA Standard 67.04, "Setpoints for Nuclear Safety-Related Instrumentation." Additionally, pre-defined limits (double-sided Operability limits and as-left limits) are determined for each instrument consistent with the guidance provided in NRC RG 1.105 and ANSI/ISA–RP67.04, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." The instrument Operability limit band in plant uncertainty calculations is synonymous with the as-found acceptance criteria band specified in ITS and is centered about the nominal equipment setting (clarified in calculations as the field trip setpoint or NTSP). The PSL Instrument Setpoint Methodology, including the method of determining instrument uncertainties, was reviewed by the NRC during the review of Extended Power Uprate (EPU) as documented in the EPU safety evaluations for License Amendments 213 (Unit 1) and 163 (Unit 2), dated July 9, 2012 and September 24, 2012, respectively (NRC ADAMS Accession Nos. ML12181A019 and ML12235A463). The NRC staff determined that the PSL Instrument Setpoint Methodology described in the EPU license request meets the requirements of 10 CFR 50, Appendix A, General Design Criterion 13 and the regulatory guidance in RG 1.105 (EPU SE pg. 92 (Unit 1) and 95 (Unit 2)). The NRC staff also determined that the application of the clarifications of RG 1.105 in Regulatory Issue Summary 2006-17 "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," is acceptable (EPU SE pg. 94 (Unit 1) and 97 (Unit 2)). | R1

The removal of these details for meeting Technical Specification requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the Allowable Values associated with the RPS Instrumentation, which are designated as the Operability limits for the required instrument Functions. Footnotes (a) and (b) in Table 3.3.1-1 are included to ensure channel performance continues to verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology consistent with the NRC guidance specified in RIS 2006-17. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA05 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 2.2-1 Functional Unit 3 provides an Allowable Value of "≥ 95% of minimum reactor coolant flow with 4 pumps operating.*" Footnote * refers to CTS LCO 3.2.5 for the minimum reactor coolant flow for 4 pumps

operating. ITS Table 3.3.1-1 Function 3 provides an Allowable Value for Reactor Coolant Flow – Low in terms of percent but does not include the detail of associated reactor coolant flow. This changes the CTS by moving the detail of what the setting in % is based on to the UFSAR.

The removal of this detail, related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the value for each of the Allowable Values. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA06 **Unit 2 only:** *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements*) CTS Table 3.3-1, Action 2 states, in part, that all functional units affected by the bypassed/tripped channel shall be placed in the bypassed/tripped condition. CTS Table 3.3-1, Action 2 provides a process instrument and Function bypass list that identifies the associated Functions that must be bypassed if the instrument becomes inoperable. ITS 3.3.1 ACTION A includes a Required Action to place the affected trip unit in trip or bypass within 1 hour. CTS LCO 3.0.2 and ITS LCO 3.0.2 establish that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. Therefore, when an instrument channel becomes inoperable, LCO 3.0.2 requires each Function affected by the inoperability shall be declared inoperable and the associated ACTIONS shall be met. The CTS process instrument and Function bypass list is not necessary in the Specifications. This information is removed from the Technical Specifications and moved to the UFSAR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that all functional units receiving an input from an inoperable channel are bypassed or tripped. ITS 3.3.1 ACTIONS for each Function include a Required Action to place the affected trip unit in trip or bypass within 1 hour. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA07 **Unit 2 only:** *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Table 2.2-1 Functional Unit 8, Allowable Values column, includes Note (3), that states, “% of the narrow range steam generator level indication.” ITS Table 3.3.1-1 Steam Generator Level – Low Function (Functions 7.a and 7.b) do not include this design detail. This changes the CTS by moving the detail of what the setting in % is based on to the UFSAR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the Allowable Value requirement for the Steam Generator Level – Low RPS Function. Also, this change is acceptable because the removed information is currently provided in the UFSAR Section 7.2 where it is adequately controlled. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 **Unit 1 only:** (*Category 2 – Relaxation of Applicability*) CTS Table 3.3-1 Functional Unit 11a Applicability requires OPERABILITY of the Wide Range Logarithmic Neutron Flux Monitor – Rate of Change of Power – High trip function in MODES 1, 2 and *. Footnote * states "with the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal." CTS Table 4.3-1 surveillance requirement applicability for Functional Unit 11 also includes a footnote * which states, "with reactor trip breaker closed." ITS 3.3.1 Function 2 requires the equivalent trip function to be OPERABLE in MODES 1 and 2. This changes the CTS by deleting the requirement for Rate of Change of Power – High trip function to be OPERABLE in MODES other than MODES 1 and 2 with the reactor trip circuit breakers (RTCBs) closed and the CEA drive system capable of CEA withdrawal.

The purpose of the requirement for the Rate of Change of Power – High trip function to be OPERABLE MODES other than MODES 1 and 2 with the RTCBs closed and the CEA drive system capable of CEA withdrawal is to ensure that a withdrawn CEA can be inserted in the event of an uncontrolled reactivity event during shutdown (i.e., MODES 3, 4, and 5). However, the Rate of Change of Power – High trip function is not assumed in any shutdown event (e.g., boron dilution event). The proposed change is consistent with the PSL Unit 2 CTS for the same function.

This change is acceptable because the uncontrolled CEA withdrawal event analysis is initiated from a MODE 2 startup condition at zero power with termination on a high-power trip signal. The high-power trip was used in the event analysis to determine the required initial overpower margin for DNBR and FCM (UFSAR Section 15.2.1). The rate of change of power trip is not credited in any accident analysis as providing primary protection for any limiting case accident, including any shutdown event. Appropriate reactor protection is provided in MODES 3, 4, and 5 when the RTCBs closed and any CEA is capable of being withdrawn by ITS LCO 3.3.2, which requires four Manual Trip channels and associated trip logic to be OPERABLE. Also, ITS LCO 3.3.11 requires two logarithmic neutron flux monitoring channels to be OPERABLE in MODES 3, 4 and 5 to monitor neutron flux power so a loss of SHUTDOWN MARGIN caused by boron dilution can be detected as an increase in flux. This change is

designated as less restrictive because the LCO requirements are applicable in fewer plant conditions than in the CTS.

L02 **Unit 1 only:** (*Category 4 – Relaxation of Required Action*) CTS Table 3.3-1 Action 2.a requires, with the number of OPERABLE channels one less than the Total Number of Channels, to place the inoperable channel in the bypassed or tripped condition but do not provide an action for more than one RPS channel inoperable. Thus, CTS 3.0.3 would be required if two or more RPS Function channels are inoperable. ITS 3.3.1 ACTION B provides an allowance for one or more Functions with two RPS trip units or associated instrument channels inoperable except for the condition when the excore channels are not calibrated with the incore detectors. ITS 3.3.1 ACTION C provides an allowance for one or more Functions with one or more power range excore channels not calibrated within the incore detectors. ITS 3.3.1 ACTION B requires one trip unit to be placed in bypass and the other trip unit must be placed in trip within 1 hour and one trip unit must be restored to OPERABLE status within 48 hours. ITS 3.3.1 ACTION C requires performing SR 3.3.1.3 (Calibrate the power range excore channels with the incore detectors) in 24 hours or restrict THERMAL POWER to ≤ 90% RTP. This changes the CTS by providing actions for the condition of two RPS Function trip units or associated instrument channels inoperable and allowing one or more excore channels not calibrated to within the incore detectors.

The RPS Function channels are configured in a two-out-of-four coincidence. The two-out-of-four RPS logic is changed to a two-out-of-three logic for a given input parameter in one channel at a time by bypassing one channel input to the logic. With one channel of protective instrumentation bypassed, the RPS Function is in two-out-of-three logic, but with another channel failed the RPS may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, RPS actuation will occur. Therefore, it is also acceptable to allow an additional channel to be inoperable. When one or more power range excore channels are discovered not calibrated with the incore detectors, time is allowed to calibrate the power range instruments instead of placing the plant in a transient with the excore power range channels not calibrated. Alternatively, THERMAL POWER may be restricted to ≤ 90% RTP to provide additional margin to the fuel thermal limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the remaining RPS instrument channels, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. This change also provides appropriate remedial actions in lieu of requiring initiation of an unnecessary plant transient per ITS LCO 3.0.3 (CTS 3.0.3). The 48-hour Completion Time of proposed Required Action B.2 is based upon operating experience, which has demonstrated that a random failure of another channel occurring during the 48 hour period is a low probability event. The 24-hour Completion Time of proposed Required Actions C.1 and C.2 is adequate to

calibrate the power range excore channels while minimizing the risk of operating in an unsafe condition. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 (*Category 4 – Relaxation of Required Action*) Unit 1 CTS 3.3.1.1 and Unit 2 CTS 3.3.1 do not explicitly provide requirements associated with the bypass removal feature associated with the following CTS Table 3.3-1 RPS Functional Units; Power Rate of Change – High (11.a), Reactor Coolant Flow – Low (3), Steam Generator Pressure – Low (6), Local Power Density – High (8), Thermal Margin/Low Pressure (9), and Loss of Load (10). However, CTS 4.3.1.1.2 and Unit 2 CTS 4.3.1.2 require channel functional and calibration testing of the channel bypass logic, which includes the bypass removal feature. Therefore, per the requirements of CTS 4.0.1 (ITS SR 3.0.1), CTS 3.3.1.1 (Unit 1) and CTS 3.3.1 (Unit 2) encompass the RPS channel bypass removal feature. If an RPS bypass removal channel is inoperable, CTS 3.0.3 is entered since there are no Conditions and Actions specified in the CTS. ITS 3.3.1 also includes Surveillances for the channel bypass removal feature. In addition, ITS 3.3.1 ACTIONS D and E are added to provide remedial measures when the bypass removal capability is inoperable to a required RPS Function channel. ITS Required Actions D.1 and E.1 require disabling the inoperable bypass channel within 1 hour when one or two bypass removal channels is inoperable. When one bypass removal channel is inoperable, ITS 3.3.1 ACTION D alternately allows placing the affected channel in bypass or trip within 1 hour and restoring the channel to OPERABLE status within 48 hours. When two bypass removal channels are inoperable, ITS 3.3.1 ACTION E alternately allows placing one channel in trip and the other channel in bypass and restoring one channel to OPERABLE status within 48 hours. This changes the CTS by adding specific Technical Specification actions when one or two bypass removal channels are inoperable to the applicable Functions.

The purpose of the proposed actions is to provide appropriate remedial actions in the event the automatic bypass removal feature is inoperable. With one or two bypass removal channels inoperable, it is permissible to continue operation with the bypass permissive removal channels failed, providing the bypass is disabled. The Completion Times are consistent with the completion times for the conditions of one or two inoperable RPS channels. This change is also consistent with the ISTS.

This change is considered acceptable because the proposed remedial actions will ensure the related RPS Functions can continue to perform their related safety function when required by the plant design basis and safety analysis by requiring the bypass removal channel to be disabled or take action consistent with the current actions for an inoperable RPS channel. This change also provides appropriate remedial actions in lieu of requiring initiation of an unnecessary plant transient per ITS LCO 3.0.3 (CTS 3.0.3). This change is designated as less restrictive because the ITS ACTIONS provide remedial actions that are less restrictive than the CTS Action requiring a plant shutdown.

L04 **Unit 2 only:** (*Category 4 – Relaxation of Required Action*) CTS Table 3.3-1 Actions 2.a and 2.b provide actions for conditions with the number of OPERABLE

channels one less than the Total Number of Channels and the condition with the number of channels OPERABLE one less than the Minimum Channels OPERABLE. ITS 3.3.1 Conditions A and B also provide Required Actions for these conditions except for the condition when the excore channels are not calibrated with the incore detectors. ITS 3.3.1 ACTION C provides an allowance for one or more Functions with one or more power range excore channels not calibrated within the incore detectors. ITS 3.3.1 ACTION C requires performing SR 3.3.1.3 (Calibrate the power range excore channels with the incore detectors) in 24 hours or restrict THERMAL POWER to ≤ 90% RTP. This changes the CTS by providing actions for the condition of one or more excore channels not calibrated to within the incore detectors.

When one or more power range excore channels is discovered not calibrated with the incore detectors, time is allowed to calibrate the power range instruments instead of placing the plant in a transient with the excore power range channels not calibrated. Alternatively, THERMAL POWER may be restricted to ≤ 90% RTP to provide additional margin to the fuel thermal limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the remaining RPS instrument channels, a reasonable time for recalibration, and the low probability of a DBA occurring during the calibration period. The 24-hour Completion Time of proposed Required Actions C.1 and C.2 is adequate to calibrate the power range excore channels while minimizing the risk of operating in an unsafe condition. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L05 **Unit 2 only:** (*Category 6 - Relaxation of Surveillance Requirement Acceptance Criteria*) CTS 4.3.1.1 and Table 4.3-1, Functional Unit 13 (Wide Range Logarithmic Neutron Flux Monitor) require, in part, a CHANNEL CALIBRATION on a Frequency in accordance with the Surveillance Frequency Control Program (SFCP). ITS SR 3.3.1.8 also requires a CHANNEL CALIBRATION but is modified by a Note stating, "Neutron detectors are excluded from CHANNEL CALIBRATION." This changes the CTS by excluding neutron detectors from the CHANNEL CALIBRATION requirement for the Logarithmic Neutron Flux Monitoring channels.

The purpose of a CHANNEL CALIBRATION is to ensure that the channel responds within the necessary range and accuracy to known values of the parameter that the channel monitors. Thus, to perform a channel calibration of a neutron flux channel would require including the neutron flux detector in the calibration. Inclusion of neutron flux detectors in the CHANNEL CALIBRATION process is impractical in power reactor applications because to do so would require subjecting the detectors to known neutron fluxes. Because of the hazards associated with exposing the neutron detectors, CTS Table 4.3-1 Note (4) excludes the detectors of Functional Unit 2 (Variable Power Level High) from CHANNEL CALIBRATION. The logarithmic neutron flux monitoring channels also use neutron detectors, which are subject to the same hazards as the detectors of

the RPS Variable Power Level - High function. The Note excluding the neutron detectors from the calibration is acceptable because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. This proposed change is consistent with historical and current NRC staff requirements as reflected in ISTS. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L06 (*Category 2 – Relaxation of Applicability*) CTS Table 3.3-1 requires the OPERABILITY of Loss of Load (Turbine Hydraulic Fluid Pressure – Low) (Unit 1 Functional Unit 10 and Unit 2 Functional Unit 15) and Low Power Density – High (Unit 1 Functional Unit 8 and Unit 2 Functional Unit 9) trip functions in MODE 1. Table 3.3-1 Notation (c) to the function(s) states, "Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is > 15% of RATED THERMAL POWER." In addition, Unit 1 Table 2.2-1 Notation (3) and Unit 2 Notation (5) state, "Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is greater than or equal to 15% of RATED THERMAL POWER." Footnote (g) to ITS Table 3.3.1-1 Functions 8 and 10 provide(s) the same allowance for bypassing and automatically removing the bypass for the Loss of Load and Local Power Density functions. In addition, ITS Footnote (g) provides similar allowances and Footnote (h) is added to clarify the Loss of Load and Local Power Density trip functions are not applicable and may be bypassed when THERMAL POWER is < 15% RTP and the functions are applicable in MODE 1 ≥ 15% RTP. This changes the CTS by relaxing the MODE 1 applicability requirement of the Loss of Load (turbine hydraulic fluid pressure – low) and Local Power Density RPS Functions to MODE 1 ≥ 15% RTP. R1

The purpose of the requirement for the Loss of Load (Turbine Hydraulic Fluid Pressure – Low) RPS Function to be OPERABLE in MODE 1 is to ensure the RPS trip function is OPERABLE in anticipation of a high pressurizer pressure RPS trip following a trip of the main turbine generator. The Loss of Load RPS trip is anticipatory for the loss of heat removal capabilities of the secondary system following a turbine trip. Thus, the trip minimizes the pressure and temperature transients on the reactor by initiating a trip before reaching the Pressurizer Pressure - High trip. The Loss of Load trip also prevents lifting the pressurizer safety valves, power operated relief valves (PORVs), and main steam safety valves (MSSVs) in the event of a main turbine generator trip. The proposed change to the Applicability is consistent with the ISTS.

The purpose of the requirement for the Local Power Density – High RPS Function to be OPERABLE in MODE 1 is to ensure excessive axial peaking will not cause fuel damage. It ensures that neither a DNBR less than the SAFETY LIMIT nor a peak linear heat rate corresponding to the temperature for fuel centerline melting will occur.

These changes are acceptable because the Loss of Load and Local Power Density RPS trips are not required to support any accident or transient event < 15% RTP. The Loss of Load and Local Power Density trips are currently allowed to be bypassed when THERMAL POWER falls below 15%. In addition, the Loss

of Load RPS and Local Power Density trips are not credited in any accident analysis as providing primary protection for any limiting case accident. Appropriate reactor protection continues to be provided below 15% RTP by other reactor trips. This change is designated as less restrictive because the Loss of Load (turbine hydraulic fluid pressure – low) and Local Power Density RPS Functions are applicable in fewer plant conditions than in the CTS.

L07 **Unit 2 only:** (*Category 7- Relaxation of Surveillance Frequency*) CTS 4.3.1.1 and Table 4.3-1, Functional Unit 15 Loss of Load (Turbine Hydraulic Fluid Pressure – Low) requires, in part, a CHANNEL FUNCTIONAL TEST on a Frequency in accordance with the Surveillance Frequency Control Program (SFCP). The Frequency specified in the SFCP is 31 days. ITS SR 3.3.1.6 requires a CHANNEL FUNCTIONAL TEST to be performed for the Loss of Load RPS Function at a Frequency of "Once within 7 days prior to each reactor startup." This changes the CTS by requiring the CHANNEL FUNCTIONAL TEST for the Loss of Load RPS Function to be performed only during a reactor startup instead of on a periodicity. R1

The purpose of a CHANNEL FUNCTIONAL TEST is to verify successful operation of the required contact(s) of a channel relay in between CHANNEL CALIBRATIONS by the verification of the change of state of the relay contacts. The Loss of Load (Turbine Hydraulic Fluid Pressure – Low) trip is anticipatory for events resulting in loss of heat removal capabilities of the secondary system following a turbine trip. The loss of load analysis provided in UFSAR Chapter 15 demonstrates that the high pressurizer pressure RPS trip Function is the primary RPS trip Function for these events. Performing a CHANNEL FUNCTIONAL TEST on a 31 day periodicity in accordance with the SFCP requires the Surveillance to be performed during reactor operation. Performing this test during reactor operation could have the adverse effect of causing a pressure perturbation in the emergency trip fluid header common to all four channels resulting in a main turbine trip and reactor trip. Therefore, it is proposed to revise the frequency to perform this test within 7 days prior to each reactor startup consistent with the Frequency specified for the same surveillance test for the Loss of Load RPS Function in Unit 1 CTS and the ISTS.

This change is considered acceptable because the CHANNEL FUNCTIONAL TEST for the Loss of Load RPS Function will still be performed at a frequency that assures the necessary quality of systems and components is maintained. A review of historical surveillance data has shown that this test consistently passes the Surveillance at the current Frequency and demonstrates that there are no failures that would invalidate an extended surveillance test interval from 31 days in accordance with the SFCP to prior to each reactor startup. Relaxation of this surveillance test interval from the current periodic frequency is offset by eliminating the need to perform the Surveillance during reactor operation, which could result in an unnecessary plant transient. Also, the proposed event based frequency will be retained in the Technical Specifications instead of maintaining a periodic frequency under licensee control. This change is designated as less restrictive because the Surveillance will be performed less frequently under the ITS than currently under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.1 Reactor Protective System (RPS) Instrumentation ~~- Operating (Analog)~~ 2

LCO 3.3.1.1
LCO 2.2.1
DOC LA04

LCO 3.3.1 Four RPS trip units and associated instrument and bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.

Applicability
Table 3.3-1

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

ACTIONS

----------NOTE----------

DOC A03

Separate Condition entry is allowed for each RPS Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action 2 A. One or more Functions with one RPS trip unit or associated instrument channel inoperable except for Condition C (excore channel not calibrated with incore detectors).	A.1 Place affected trip unit in bypass or trip. AND A.2.1 Restore channel to OPERABLE status. OR A.2.2 ~~[~~Place affected trip unit in trip.	1 hour ~~[~~48~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ 48 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~ 3

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L02 B. One or more Functions with two RPS trip units or associated instrument channels inoperable except for Condition C (excore channel not calibrated with incore detectors).	B.1 Place one trip unit in bypass and place the other trip unit in trip. AND B.2 Restore one trip unit to OPERABLE status.	1 hour ~~[~~48~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ (3)
DOC L02 C. One or more Functions with one or more power range excore channels not calibrated with the incore detectors.	C.1 Perform SR 3.3.1.3. OR C.2 Restrict THERMAL POWER to ≤ 90% RTP.	24 hours 24 hours
DOC L03 D. One or more Functions with one automatic bypass removal channel inoperable.	D.1 Disable bypass channel. OR D.2.1 Place affected trip units in bypass or trip. AND D.2.2.1 Restore bypass removal channel and affected trip units to OPERABLE status. OR	1 hour 1 hour ~~[~~48~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ (3)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	D.2.2.2 ~~[~~Place affected trip units in trip.	48 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
DOC L03 E. One or more Functions with two automatic bypass removal channels inoperable.	E.1 Disable bypass channels. OR E.2.1 ~~[~~Place one affected trip unit in bypass and place the other in trip for each affected trip Function. AND E.2.2 Restore one automatic bypass removal channel and the associated trip unit to OPERABLE status for each affected trip Function.	1 hour 1 hour ~~[~~48~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
DOC A08 F. Required Action and associated Completion Time not met for ~~Axial Power Distribution~~ and Loss of Load Trip Functions. (Local Power Density)	F.1 Reduce THERMAL POWER to < 15% RTP.	6 hours
DOC A08 G. Required Action and associated Completion Time not met except for ~~Axial Power Distribution~~ or Loss of Load Trip Functions. (Local Power Density)	G.1 Be in MODE 3.	6 hours

CTS

SURVEILLANCE REQUIREMENTS

----------NOTE----------

4.3.1.1.1 Refer to Table 3.3.1-1 to determine which SR shall be performed for each RPS Function.

	SURVEILLANCE		FREQUENCY	
4.3.1.1.1 Table 4.3-1 Functional Units 2a, 2b, 3, 4, 5, 6, 7, 8, 9, 9a and 11	SR 3.3.1.1	Perform a CHANNEL CHECK of each RPS instrument channel except Loss of Load.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3
4.3.1.1.1 Functional Units 2a and 2b Table 4.3-1 Notation (2) DOC M04 Table 4.3-1 Notation (4)	SR 3.3.1.2	----------NOTES---------- 1. Not required to be performed until 12 hours after THERMAL POWER is ≥ ~~[20]~~% RTP. (15) 2. The ~~daily~~ calibration may be suspended during PHYSICS TESTS, provided the calibration is performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau. --- Perform calibration (heat balance only) and adjust the excore power range and ΔT power channels to agree with calorimetric calculation ~~if the absolute difference is ≥ [1.5]%~~.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3 1 3

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3 4.3.1.1.1 Functional Units 2, 8, 9 and 11 Table 4.3-1 Notation (3) DOC M05	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER is ≥ ~~[~~20~~]~~% RTP. 15 ---------- Calibrate the power range excore channels using the incore detectors.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 3.3.1.4 4.3.1.1.1 Functional Units 2a, 2b, 3, 4, 5, 6, 7, 8, 9 and 9a Table 4.3-1 Notations (6) and (7)	Perform a CHANNEL FUNCTIONAL TEST of each RPS channel except Loss of Load and Power Rate of Change.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 3
SR 3.3.1.5 4.3.1.1.1 Functional Unit 2, 8, 9 and 11 Table 4.3-1 Notation 5	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform a CHANNEL CALIBRATION on excore power range channels.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.3.1.1.1 Functional Units 10 and 11 Table 4.3-1 Notation (1)	SR 3.3.1.6	Perform a CHANNEL FUNCTIONAL TEST of each Power Rate of Change channel and each Loss of Load functional unit.	Once within 7 days prior to each reactor startup	R1
4.3.1.1.2 Functional Units 2, 3, 8, 9, 9a, and 10	SR 3.3.1.7	Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function.	~~Once within 92 days prior to each reactor startup~~ In accordance with the Surveillance Frequency Control Program	(1)
4.3.1.1.2 Functional Units 2, 3, 4, 5, 6, 7, 8, 9, 9a, 10 and 11 Table 4.3-1 Notation (5) DOC M03	SR 3.3.1.8	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform a CHANNEL CALIBRATION of each RPS instrument channel, including bypass removal functions.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	(3)
4.3.1.1.3 Functional Units 2, 3, 4, 5, 6, 7, 8, 9, 9a, and 11 Notation (5)	SR 3.3.1.9	----------NOTE---------- Neutron detectors are excluded. ---------- Verify RPS RESPONSE TIME is within limits.	~~[[18] months on a STAGGERED TEST BASIS~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	R1 (3)

CTS

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
Functional Unit 2 Table 2.2-1 Notation (1) Table 3.3.1-1 Notation (a)	1. Variable ~~High Power Trip~~ Power Level – High[c]	1, 2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4[a][b] SR 3.3.1.5[a][b] SR 3.3.1.8[a][b] SR 3.3.1.9	≤ ~~[10]~~ 9.61% RTP above current THERMAL POWER but not < ~~[30]~~ 15% RTP nor > ~~[~~107~~]~~ % RTP
Functional Unit 11a Table 2.2-1 Notation (4) Table 3.3-1 Notation (d) and ** DOC L01	2. Wide Range Logarithmic Neutron Flux Monitor Power Rate of Change - High[d][e]	1, 2	SR 3.3.1.1 SR 3.3.1.6[a][b] SR 3.3.1.7 SR 3.3.1.8[a][b]	≤ ~~[2.6]~~ 2.49 dpm
Functional Unit 3 Table 2.2-1 Notation (1) Table 3.3-1 Notation (a)	3. Reactor Coolant Flow - Low[c]	1, 2	SR 3.3.1.1 SR 3.3.1.4[a][b] SR 3.3.1.7 SR 3.3.1.8[a][b] SR 3.3.1.9	≥ ~~[~~95~~]~~%
Functional Unit 4	4. Pressurizer Pressure - High	1, 2	SR 3.3.1.1 SR 3.3.1.4[a][b] SR 3.3.1.8[a][b] SR 3.3.1.9	≤ ~~[~~2400~~]~~ psia
Functional Unit 5	5. Containment Pressure - High	1, 2	~~[~~SR 3.3.1.1~~]~~ SR 3.3.1.4[a][b] SR 3.3.1.8[a][b] SR 3.3.1.9	≤ ~~[4.0]~~ 3.3 psig

Table 4.3-1 Footnote (6) DOC M01 (a) If the as-found instrument channel setpoint is outside its predefined as-found acceptance criteria band ~~tolerance~~, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

Table 4.3-1 Footnote (7) DOC M02 (b) The instrument channel setpoint shall be reset to a value that is within the as-left acceptance criteria band ~~tolerance~~ around the Nominal ~~Limiting~~ Trip Setpoint (~~L~~NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the ~~L~~NTSP are acceptable provided that the as-found and as-left ~~tolerances~~ apply to the actual setpoint implemented in the Surveillance procedures ~~(Nominal Trip Setpoint)~~ to confirm channel performance. The ~~L~~NTSP and the methodologies used to determine the as-found and as-left ~~tolerances~~ are specified in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~ the Technical Requirements Manual and Section 7.2 of the Updated Safety Analysis Report, respectively.

Table 2.2-1 Notation (4) Table 3.3.1-1 Notation (d) (~~d~~)(d) Trip may be bypassed when THERMAL POWER is < ~~[~~1E-4~~]~~% RTP or > ~~[13]~~ 15% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ ~~[~~1E-4~~]~~% RTP and ≤ ~~[13]~~ 15% RTP.

Table 2.2-1 Notation (1) Table 3.3.1-1 Notation (a) (~~d~~)(c) Trips may be bypassed when THERMAL POWER is < ~~[~~1E-4~~]~~%. Bypass shall be automatically removed when THERMAL POWER is ≥ ~~[~~1E-4~~]~~% RTP. ~~During testing pursuant to LCO 3.4.17, RCS Loops - Test Exceptions, trips may be bypassed below 5% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ 5% RTP.~~

Table 2.2-1 Notation ** (e) Trip not applicable when THERMAL POWER is > 15% RTP.

Reorder

CTS

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	
Functional Unit 6 Table 2.2-1 Notation (2) Table 3.3-1 Notation (b)	6. Steam Generator Pressure - Low~~(e)~~ (f)	1, 2	SR 3.3.1.1 SR 3.3.1.4(a)(b) SR 3.3.1.7 SR 3.3.1.8(a)(b) SR 3.3.1.9	≥ ~~[685]~~ 600 psia	
Functional Unit 7	7a. Steam Generator A Level - Low	1, 2	SR 3.3.1.1 SR 3.3.1.4(a)(b) SR 3.3.1.8(a)(b) SR 3.3.1.9	≥ ~~[24.7]~~ 35%	3
Functional Unit 7	7b. Steam Generator B Level - Low	1, 2	SR 3.3.1.1 SR 3.3.1.4(a)(b) SR 3.3.1.8(a)(b) SR 3.3.1.9	≥ ~~[24.7]~~ 35%	
Functional Unit 8 Table 2.2-1 Notation (3) Table 3.3-1 Notation (c) DOC L06	~~[~~8. ~~Axial Power Distribution~~ Local Power Density - High	1~~(f)~~ (g) (h)	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4(a)(b) SR 3.3.1.5(a)(b) SR 3.3.1.7 SR 3.3.1.8(a)(b) SR 3.3.1.9	Figure 3.3.1-3~~]~~	1

Table 4.3-1 Footnote (6) DOC M01 (a) If the as-found instrument channel setpoint is outside its predefined as-found acceptance criteria band ~~tolerance~~, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

Table 4.3-1 Footnote (7) DOC M02 (b) The instrument channel setpoint shall be reset to a value that is within the as-left acceptance criteria band ~~tolerance~~ around the Nominal ~~Limiting~~ Trip Setpoint (~~L~~TSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. N Setpoints more conservative than the ~~L~~TSP are acceptable provided that the as-found and as-left ~~tolerances~~ apply to the actual setpoint implemented in the Surveillance procedures ~~(Nominal Trip Setpoint)~~ to confirm channel performance. The ~~L~~TSP and the methodologies used to determine the as-found and as-left ~~tolerances~~ are specified in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~. the Technical Requirements Manual and Section 7.2 of the Updated Safety Analysis Report, respectively

4

Table 2.2-1 Notation (2) Table 3.3-1 Notation (b) (~~e~~) (f) Trip may be bypassed when steam generator pressure is < ~~[785]~~ 685 psig. Bypass shall be automatically removed when steam generator pressure is ≥ ~~[785]~~ 685 psig.

Table 2.2-1 Notation (3) Table 3.3-1 Notation (c) (~~f~~) (g) Trip is not applicable and may be bypassed when THERMAL POWER is < ~~[~~15~~]~~% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ ~~[~~15~~]~~% RTP.

DOC L06 (~~g~~) (h) Trip is only applicable in MODE 1 ≥ ~~[~~15~~]~~% RTP.

3

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Table 3.3.1-1 (page 3 of 3)
Reactor Protective System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
Functional Unit 9 Table 2.2-1 Notation (1) Table 3.3-1 Notation (a) DOC M06	9a. Thermal Margin/Low Pressure (TM/LP)(c)(~~d~~)	1, 2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4(a)(b) SR 3.3.1.5(a)(b) SR 3.3.1.7 [SR 3.3.1.8](a)(b) SR 3.3.1.9	Figures 3.3.1-1 and 3.3. 1-2 and ≥ 1887 psia (1)
Functional Unit 9a Table 2.2-1 Notation (1) Table 3.3-1 Notation (a)	[9b. Steam Generator Pressure Difference(c)(~~d~~)	1, 2	SR 3.3.1.1 SR 3.3.1.4(a)(b) SR 3.3.1.8(a)(b) SR 3.3.1.9	≤ [135] psid] (3)
Functional Unit 10 Table 2.2-1 Notation (3) Table 3.3-1 Notation (c) DOC L06	10. Loss of Load (turbine ~~stop valve control oil pressure~~ hydraulic fluid pressure - low)	1(h)(~~f~~) (g)	SR 3.3.1.6(a)(b) SR 3.3.1.7 SR 3.3.1.8(a)(b)	≥ [800] psig (3)

Table 4.3-1 Footnote (6) DOC M01 (a) If the as-found instrument channel setpoint is outside its predefined as-found acceptance criteria band ~~tolerance~~, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. acceptance criteria bands

Table 4.3-1 Footnote (7) DOC M02 (b) The instrument channel setpoint shall be reset to a value that is within the as-left acceptance criteria band ~~tolerance~~ around the Nominal ~~Limiting~~ Trip Setpoint (~~L~~TSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the ~~L~~TSP are acceptable provided that the as-found and as-left ~~tolerances~~ apply to the actual setpoint implemented in the Surveillance procedures ~~(Nominal Trip Setpoint)~~ to confirm channel performance. The ~~L~~TSP and the methodologies used to determine the as-found and as-left ~~tolerances~~ are specified in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~ the Technical Requirements Manual and Section 7.2 of the Updated Safety Analysis Report, respectively. N (4) (3)

Table 2.2-1 Notation (1) Table 3.3-1 Notation (a) (c) (~~d~~) Trips may be bypassed when THERMAL POWER is < [1~~E-4~~]%. Bypass shall be automatically removed when THERMAL POWER is ≥ [1~~E-4~~]% RTP. ~~During testing pursuant to LCO 3.4.17, trips may be bypassed below 5% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ 5% RTP.~~ (5)

Table 2.2-1 Notation (3) Table 3.3-1 Notation (c) (g) (~~f~~) Trip is not applicable and may be bypassed when THERMAL POWER is < [15]% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ [15]% RTP. (3)

DOC L06 (h) (~~g~~) Trip is only applicable in MODE 1 ≥ [15]% RTP.

CTS

Replace with CTS Figure 2.2-3 1

Figure 2.2-3

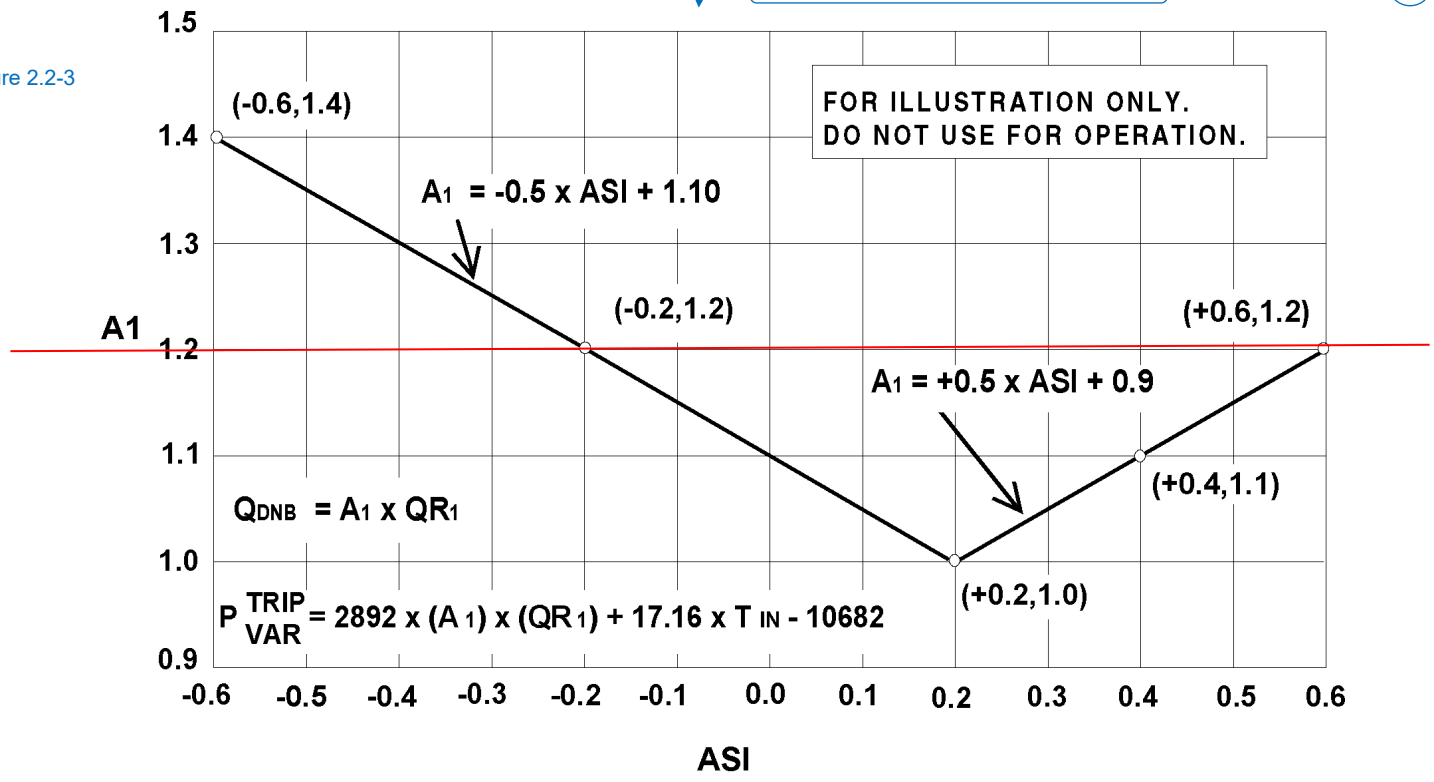

Figure 3.3.1-1 (page 1 of 1)
Thermal Margin/Low Pressure Trip Setpoint: ASI vs A1

CTS

Replace with CTS Figure 2.2-4

1

Figure 2.2-4

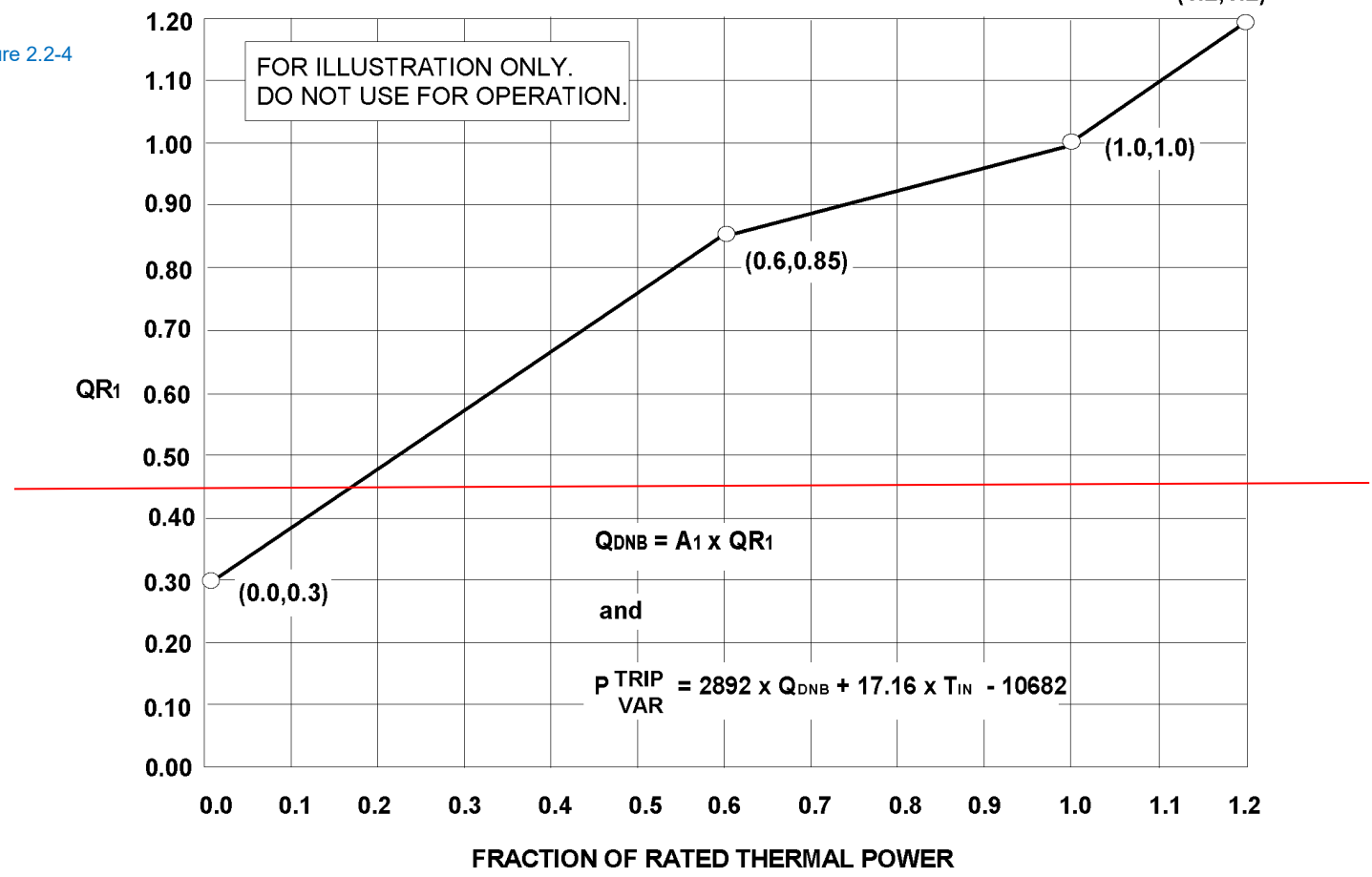

Figure 3.3.1-2 (page 1 of 1)
Thermal Margin/Low Pressure Trip Setpoint: Fraction of RTP vs QR 1

Insert CTS Figure 2.2-1

CTS

Replace with CTS Figure 2.2-2 1

Figure 2.2-2

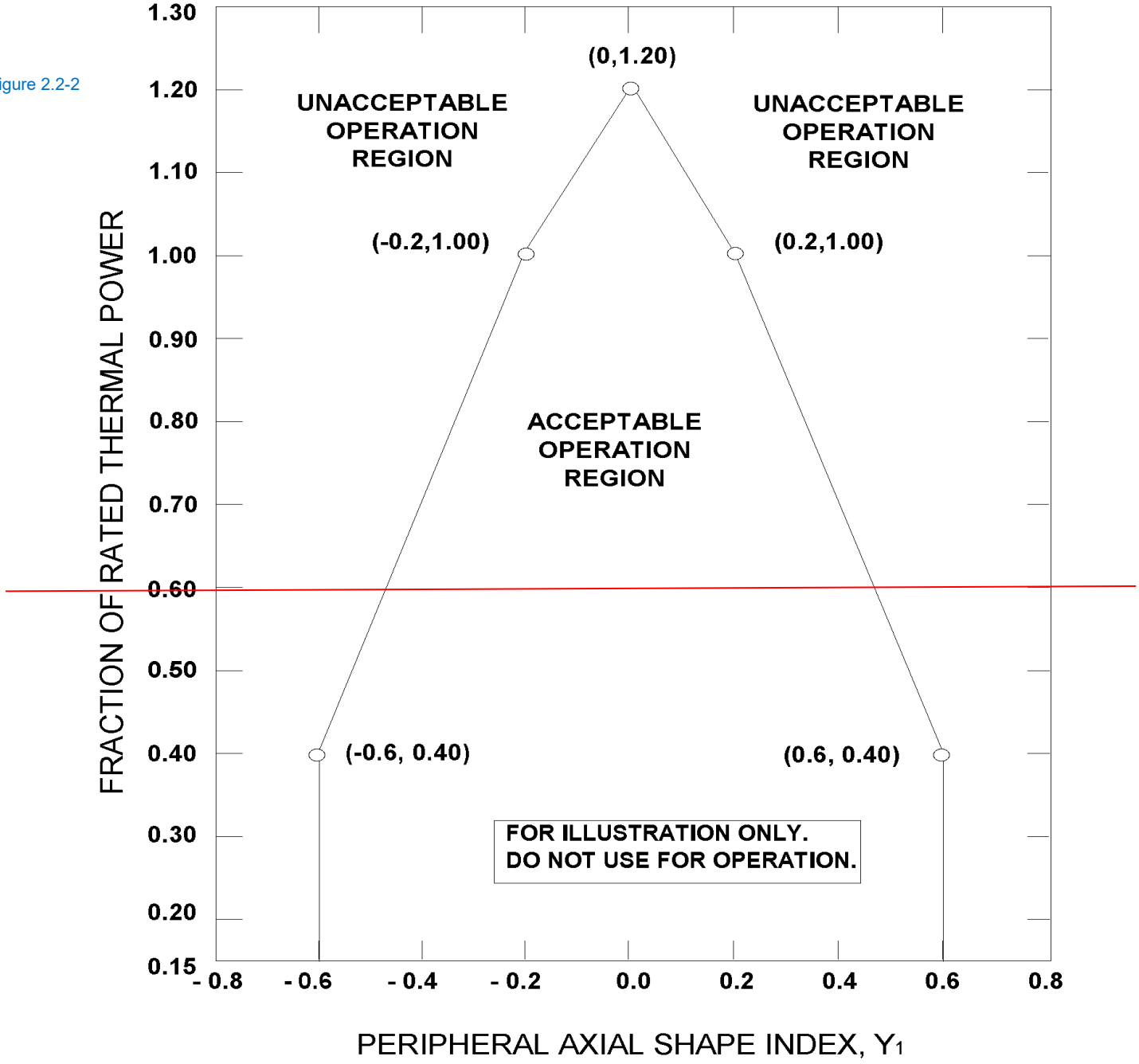

Figure 3.3.1-3 (page 1 of 1)
Peripheral Axial Shape Index, Y_1 vs Fraction of RTP

CTS

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.1 Reactor Protective System (RPS) Instrumentation ~~– Operating (Analog)~~

LCO 2.2.1
LCO 3.3.1
DOC LA04

LCO 3.3.1 Four RPS trip units and associated instrument and bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.

Applicability
Table 3.3-1

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

ACTIONS

----------NOTE----------
DOC A03 Separate Condition entry is allowed for each RPS Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action 2.a DOC A10 A. One or more Functions with one RPS trip unit or associated instrument channel inoperable except for Condition C (excore channel not calibrated with incore detectors). ----------NOTE---------- Required Action A.2 must be completed whenever this Condition is entered. ---	A.1 Place affected trip unit in bypass or trip. AND A.2~~.1~~ Restore channel to OPERABLE status. ~~OR~~ ~~A.2.2 [Place affected trip unit in trip.~~	1 hour Prior to entering MODE 2 following next MODE 5 entry. ~~[48] hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ ~~48 hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~

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CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action 2.b B. One or more Functions with two RPS trip units or associated instrument channels inoperable except for Condition C (excore channel not calibrated with incore detectors).	B.1 Place one trip unit in bypass and place the other trip unit in trip. ~~AND~~ ~~B.2 Restore one trip unit to OPERABLE status.~~	1 hour ~~[48] hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ 7
DOC L02 C. One or more Functions with one or more power range excore channels not calibrated with the incore detectors.	C.1 Perform SR 3.3.1.3. OR C.2 Restrict THERMAL POWER to ≤ 90% RTP.	24 hours 24 hours
DOC L03 D. One or more Functions with one automatic bypass removal channel inoperable.	D.1 Disable bypass channel. OR D.2.1 Place affected trip units in bypass or trip. AND D.2.2.1 Restore bypass removal channel and affected trip units to OPERABLE status. OR	1 hour 1 hour ~~[~~48~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ 3

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	D.2.2.2 ~~[~~Place affected trip units in trip.	48 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~ 3
DOC L03 E. One or more Functions with two automatic bypass removal channels inoperable.	E.1 Disable bypass channels. OR E.2.1 ~~[~~Place one affected trip unit in bypass and place the other in trip for each affected trip Function. AND E.2.2 Restore one automatic bypass removal channel and the associated trip unit to OPERABLE status for each affected trip Function.	1 hour 1 hour ~~[~~48~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~ 3
DOC A08 F. Required Action and associated Completion Time not met for ~~Axial Power Distribution~~ and Loss of Load Trip Functions. Local Power Density	F.1 Reduce THERMAL POWER to < 15% RTP.	6 hours 1
DOC A08 G. Required Action and associated Completion Time not met except for ~~Axial Power Distribution~~ or Loss of Load Trip Functions. Local Power Density	G.1 Be in MODE 3.	6 hours 1

CTS

SURVEILLANCE REQUIREMENTS

----------NOTE----------

4.3.1.1 Refer to Table 3.3.1-1 to determine which SR shall be performed for each RPS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1 4.3.1.1 Table 4.3-1 Functional Units 2a, 2b, 3, 4, 5, 6, 7, 8, 9, 13, and 14	Perform a CHANNEL CHECK of each RPS instrument channel except Loss of Load.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)
SR 3.3.1.2 4.3.1.1 Table 4.3-1 Functional Units 2a and 2b Notation (2) Notation (5) DOC M04	----------NOTES---------- 1. Not required to be performed until 12 hours after THERMAL POWER is ≥ ~~[20]~~% RTP. (15) (3) 2. The ~~daily~~ calibration may be suspended during PHYSICS TESTS, provided the calibration is performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau. (1) | R1 --- Perform calibration (heat balance only) and adjust the excore power range and ΔT power channels to agree with calorimetric calculation ~~if the absolute difference is ≥ [1.5]%~~.	~~[~~ 24 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.3 4.3.1.1 Table 4.3-1 Functional Unit 2, 4, 9, and 13a Notation (3) DOC M05	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER is ≥ ~~[~~20~~]~~% RTP. 15 ---------- Calibrate the power range excore channels using the incore detectors.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 3.3.1.4 4.3.1.1 Table 4.3-1 Functional Units 2a, 2b, 3, 4, 5, 6, 7, 8, 9, 10 and 14	Perform a CHANNEL FUNCTIONAL TEST of each RPS channel except Loss of Load and Power Rate of Change.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 3
SR 3.3.1.5 4.3.1.1 Table 4.3-1 Functional Unit 2, 4, 9, and 13a Notation (4)	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform a CHANNEL CALIBRATION on excore power range channels.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.3.1.1 Table 4.3-1 Functional Units 13 and 15 DOC L07	SR 3.3.1.6	Perform a CHANNEL FUNCTIONAL TEST of each Power Rate of Change channel and each Loss of Load functional unit.	Once within 7 days prior to each reactor startup	R1
4.3.1.2 Table 4.3-1 Functional Units 2, 4, 7, 9, 14 and 15	SR 3.3.1.7	Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function.	Once within 92 days prior to each reactor startup	R1
4.3.1.2 Functional Units 2, 3, 4, 5, 6, 7, 8, 9, 10, 13, 14 and 15 Notation (4) DOC M03 DOC L04	SR 3.3.1.8	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform a CHANNEL CALIBRATION of each RPS instrument channel, including bypass removal functions.	[~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	(3)
4.3.1.3 Functional Units 2, 3, 4, 5, 6, 7, 8, 9, 13, 14 and 15 Notation (4)	SR 3.3.1.9	----------NOTE---------- Neutron detectors are excluded. ---------- Verify RPS RESPONSE TIME is within limits.	~~[[18] months on a STAGGERED TEST BASIS~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	R1 (3)

CTS

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
Functional Unit 2 Table 3.3-1 Footnote a Table 2.2-1 Footnote (1)	1. Variable ~~High Power Trip~~ (Power Level – High[c])	1, 2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4[a][b] SR 3.3.1.5[a][b] SR 3.3.1.8[a][b] SR 3.3.1.9	≤ ~~[10]~~% RTP above current THERMAL POWER but not < ~~[30]~~% RTP nor > ~~[~~107~~]~~ % RTP (9.61; 15)
Functional Unit 13 Table 3.3-1 Footnotes ** and e Table 2.2-1 Footnote (4)	2. Power Rate of Change - High[d] (Wide Range Logarithmic Neutron Flux Monitor)	1, 2 (e)	SR 3.3.1.1 SR 3.3.1.6[a][b] SR 3.3.1.7 SR 3.3.1.8[a][b]	≤ ~~[2.6]~~ dpm (2.49)
Functional Unit 14 Table 3.3-1 Footnote a Table 2.2-1 Footnote (1)	3. Reactor Coolant Flow - Low[d] (c)	1, 2	SR 3.3.1.1 SR 3.3.1.4[a][b] SR 3.3.1.7 SR 3.3.1.8[a][b] SR 3.3.1.9	≥ ~~[95]~~% (94.9)
Functional Unit 3	4. Pressurizer Pressure - High	1, 2	SR 3.3.1.1 SR 3.3.1.4[a][b] SR 3.3.1.8[a][b] SR 3.3.1.9	≤ ~~[2400]~~ psia (2374)
Functional Unit 5	5. Containment Pressure - High	1, 2	~~[~~SR 3.3.1.1~~]~~ SR 3.3.1.4[a][b] SR 3.3.1.8[a][b] SR 3.3.1.9	≤ ~~[4.0]~~ psig (3.1)

(1) (3)

(a) If the as-found instrument channel setpoint is outside its predefined as-found ~~tolerance~~, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. (acceptance criteria band; acceptance criteria bands) — Table 4.3-1 Footnote (8) DOC M01

(b) The instrument channel setpoint shall be reset to a value that is within the as-left ~~tolerance~~ around the ~~Limiting~~ Trip Setpoint (~~L~~TSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the ~~L~~TSP are acceptable provided that the as-found and as-left ~~tolerances~~ apply to the actual setpoint implemented in the Surveillance procedures ~~(Nominal Trip Setpoint)~~ to confirm channel performance. The ~~L~~TSP and the methodologies used to determine the as-found and as-left ~~tolerances~~ are specified in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~. (acceptance criteria band; Nominal; N; the Technical Requirements Manual and Section 7.2 of the Updated Safety Analysis Report, respectively) — Table 4.3-1 Footnote (9) DOC M02

(4) (3)

(~~c~~d) Trip may be bypassed when THERMAL POWER is < ~~[~~1E-4~~]~~% RTP or > ~~[13]~~% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ ~~[~~1E-4~~]~~% RTP and ≤ ~~[13]~~% RTP. (15) — Table 3.3-1 Footnote e Table 2.2-1 Footnote (4)

(3)

(~~d~~c) Trips may be bypassed when THERMAL POWER is < ~~[1E-4]~~% (0.5). Bypass shall be automatically removed when THERMAL POWER is ≥ ~~[1E-4]~~% RTP (0.5). ~~During testing pursuant to LCO 3.4.17, RCS Loops – Test Exceptions, trips may be bypassed below 5% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ 5% RTP.~~ — Table 3.3-1 Footnote e Table 2.2-1 Footnote (1)

(5)

(e) Trip not applicable when THERMAL POWER is > 15% RTP. — Table 2.2-1 Notation **

(1)

Reorder

CTS

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	
Functional Unit 6 Table 3.3-1 Footnote b Table 2.2-1 Footnote (2)	6. Steam Generator Pressure - Low~~(e)~~ (f)	1, 2	SR 3.3.1.1 SR 3.3.1.4(a)(b) SR 3.3.1.7 SR 3.3.1.8(a)(b) SR 3.3.1.9	≥ ~~[685]~~ 621 psia	
Functional Unit 8	7a. Steam Generator A Level - Low	1, 2	SR 3.3.1.1 SR 3.3.1.4(a)(b) SR 3.3.1.8(a)(b) SR 3.3.1.9	≥ ~~[24.7]~~ 35 %	3
Functional Unit 8	7b. Steam Generator B Level - Low	1, 2	SR 3.3.1.1 SR 3.3.1.4(a)(b) SR 3.3.1.8(a)(b) SR 3.3.1.9	≥ ~~[24.7]~~ 35 %	
Functional Unit 9 Table 3.3-1 Footnote c Table 2.2-1 Footnote (5) DOC L06	~~[~~8. ~~Axial Power Distribution~~ Local Power Density - High	1~~(f)~~(g) (h)	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4(a)(b) SR 3.3.1.5(a)(b) SR 3.3.1.7 SR 3.3.1.8(a)(b) SR 3.3.1.9	Figure 3.3.1-3~~]~~	1

Table 4.3-1 Footnote (8) DOC M01 (a) If the as-found instrument channel setpoint is outside its predefined as-found acceptance criteria band ~~tolerance~~, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. acceptance criteria bands

Table 4.3-1 Footnote (9) DOC M02 (b) The instrument channel setpoint shall be reset to a value that is within the as-left acceptance criteria band ~~tolerance~~ around the Nominal ~~Limiting~~ Trip Setpoint (~~L~~TSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. N Setpoints more conservative than the ~~L~~TSP are acceptable provided that the as-found and as-left ~~tolerances~~ apply to the actual setpoint implemented in the Surveillance procedures ~~(Nominal Trip Setpoint)~~ to confirm channel performance. The ~~L~~TSP and the methodologies used to determine the as-found and as-left ~~tolerances~~ are specified in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~. the Technical Requirements Manual and Section 7.2 of the Updated Safety Analysis Report, respectively 4

Table 3.3-1 Footnote b Table 2.2-1 Footnote (2) (f) (~~e~~) Trip may be bypassed when steam generator pressure is < ~~[785]~~ 705 psig. Bypass shall be automatically removed when steam generator pressure is ≥ ~~[785]~~ psig.

Table 3.3-1 Footnote c Table 2.2-1 Footnote (5) DOC L06 (g) (~~f~~) Trip is not applicable and may be bypassed when THERMAL POWER is < ~~[~~15~~]~~% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ ~~[~~15~~]~~% RTP.

(h) (~~g~~) Trip is only applicable in MODE 1 ≥ ~~[~~15~~]~~% RTP. 3

CTS

Table 3.3.1-1 (page 3 of 3)
Reactor Protective System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
Functional Unit 4 Table 3.3-1 Footnote a Table 2.2-1 Footnote 1	9a. Thermal Margin/Low Pressure (TM/LP)~~(d)~~ (c)	1, 2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4(a)(b) SR 3.3.1.5(a)(b) SR 3.3.1.7 ~~[~~SR 3.3.1.8~~]~~(a)(b) SR 3.3.1.9	Figures 3. 3.1-1 and 3.3.1-2 and ≥ 1900 psia (1)
Functional Unit 7 Table 3.3-1 Footnote a Table 2.2-1 Footnote (1)	9b. Steam Generator Pressure Difference~~(d)~~ (c)	1, 2	SR 3.3.1.1 SR 3.3.1.4(a)(b) SR 3.3.1.8(a)(b) SR 3.3.1.9	≤ ~~[~~135~~]~~ psid~~]~~ (2)
Functional Unit 15 Table 3.3-1 Footnote c Table 2.2-1 Footnote (5) DOC L06	10. Loss of Load (turbine ~~stop valve control oil pressure~~) (hydraulic fluid pressure - low)	1 ~~(f)~~ (g) (h)	SR 3.3.1.6(a)(b) SR 3.3.1.7 SR 3.3.1.8(a)(b)	≥ ~~[~~800~~]~~ psig

(3)

Table 4.3-1 Footnote (8) DOC M01 (a) If the as-found (instrument) channel setpoint is outside its predefined as-found (acceptance criteria band) ~~tolerance~~, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

Table 4.3-1 Footnote (9) DOC M02 (b) The instrument channel setpoint shall be reset to a value that is within the as-left (acceptance criteria band) ~~tolerance~~ around the (Nominal) ~~Limiting~~ Trip Setpoint (~~L~~TSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. (N) Setpoints more conservative than the ~~L~~TSP are acceptable provided that the as-found and as-left ~~tolerances~~ apply to the actual setpoint implemented in the Surveillance procedures ~~(Nominal Trip Setpoint)~~ to confirm channel performance. The ~~L~~TSP and the methodologies used to determine the as-found and as-left ~~tolerances~~ are specified in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~. (the Technical Requirements Manual and Section 7.2 of the Updated Safety Analysis Report, respectively)

(acceptance criteria bands) (4)

Table 3.3-1 Footnote a Table 2.2-1 Footnote (1) DOC A09 (~~d~~) (c) Trips may be bypassed when THERMAL POWER is < ~~[~~1E-4~~]~~%. Bypass shall be automatically removed when THERMAL POWER is ≥ ~~[~~1E-4~~]~~% RTP. ~~During testing pursuant to LCO 3.4.17, trips may be bypassed below 5% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ 5% RTP.~~ (0.5) (5)

Table 3.3-1 Footnote c Table 2.2-1 Footnote (5) (~~f~~) (g) Trip is not applicable and may be bypassed when THERMAL POWER is < ~~[~~15~~]~~% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ ~~[~~15~~]~~% RTP.

DOC L06 (~~g~~) (h) Trip is only applicable in MODE 1 ≥ ~~[~~15~~]~~% RTP.

(3)

CTS

Replace with CTS Figure 2.2-3 1

Figure 2.2-3

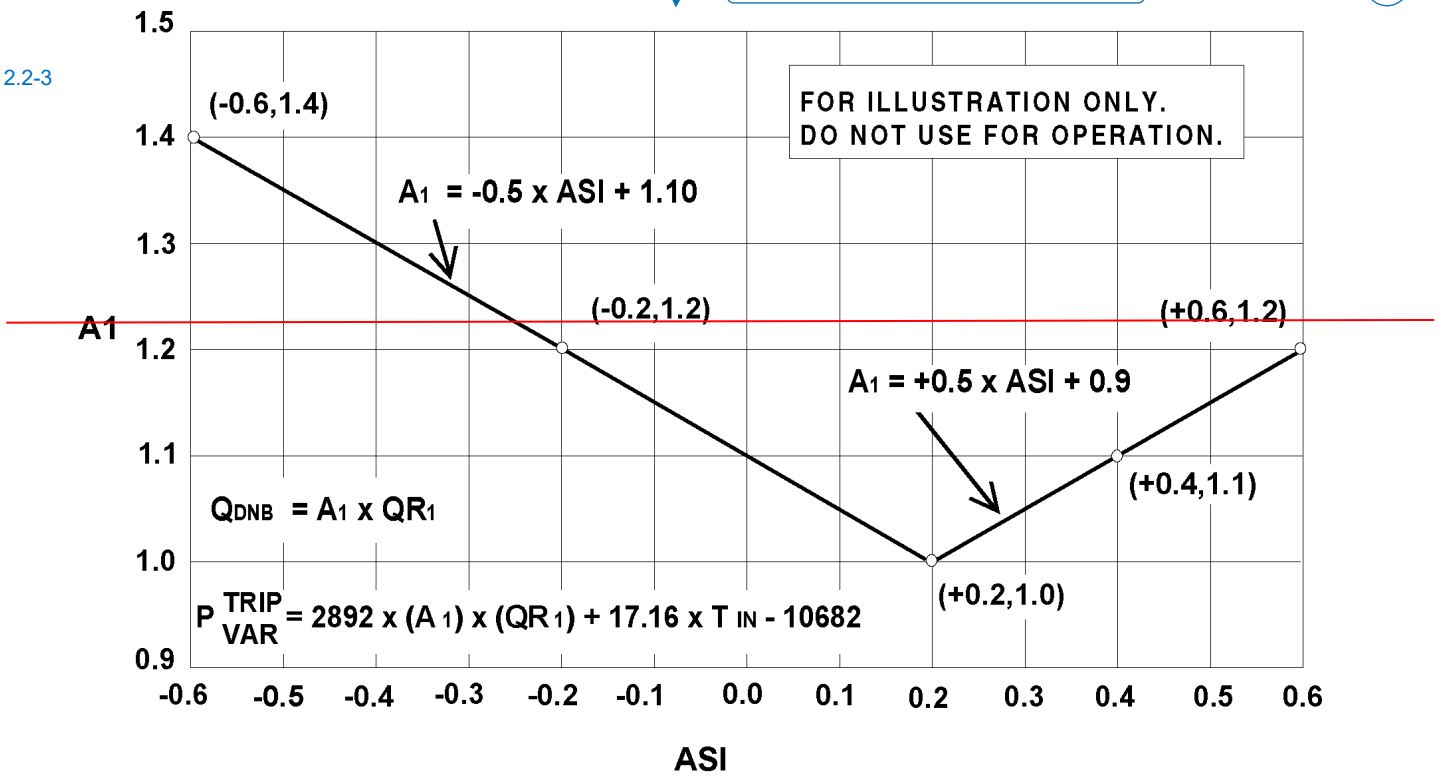

Figure 3.3.1-1 (page 1 of 1)
Thermal Margin/Low Pressure Trip Setpoint: ASI vs A1

CTS

Replace with CTS Figure 2.2-4

1

Figure 2.2-4

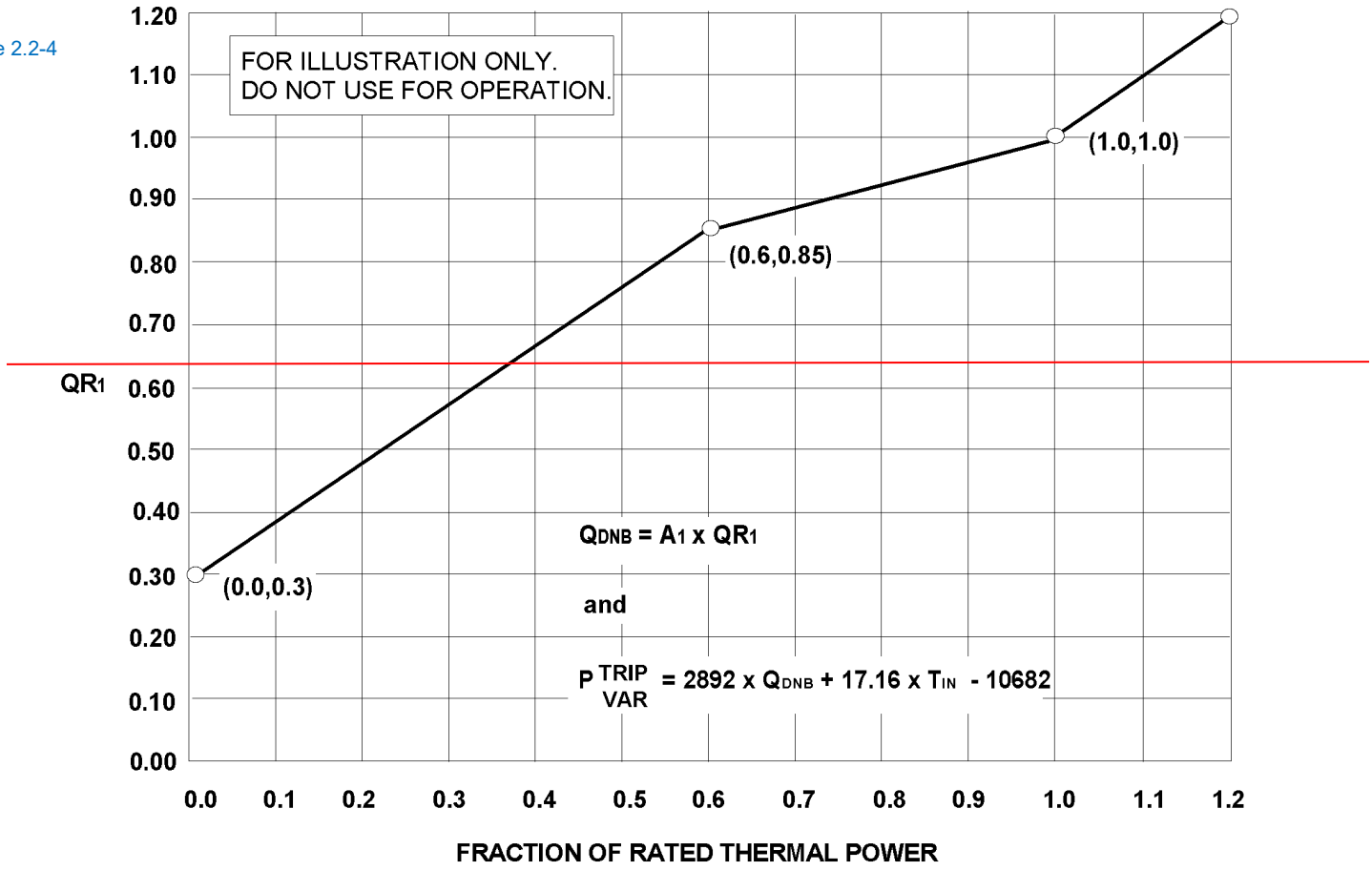

Figure 3.3.1-2 (page 1 of 1)
Thermal Margin/Low Pressure Trip Setpoint: Fraction of RTP vs QR 1

Insert CTS Figure 2.2-1

CTS

Replace with CTS Figure 2.2-2 1

Figure 2.2-2

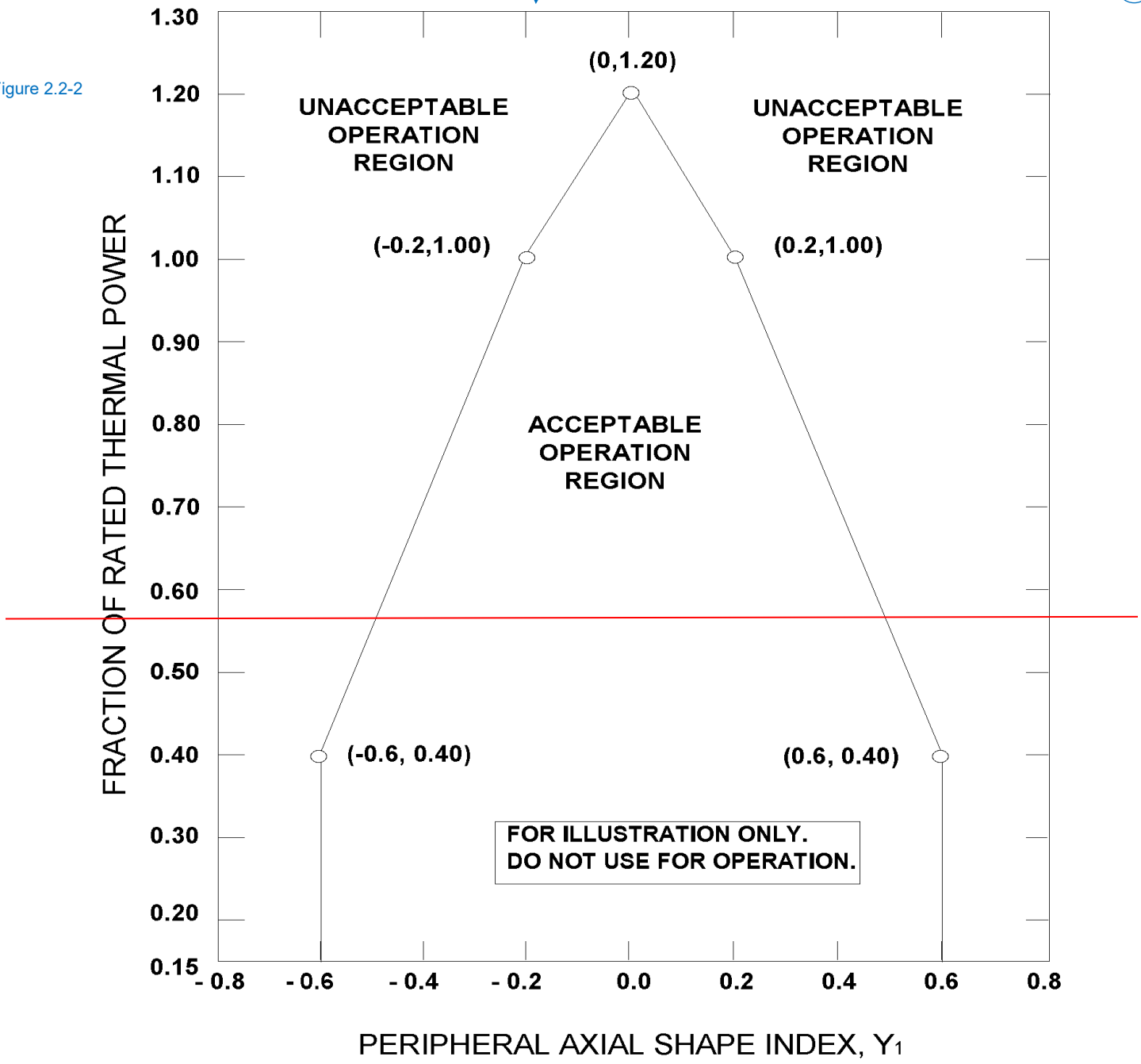

Figure 3.3.1-3 (page 1 of 1)
Peripheral Axial Shape Index, Y_1 vs Fraction of RTP

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.1, REACTOR PROTECTIVE SYSTEM (RPS) INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The heading for ISTS 3.3.1 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

4. PSL does not use the terms “as-found tolerance” and “Limiting Trip Setpoint (LTSP)” in plant specific instrument calculations or the Updated Final Safety Analysis Report. Therefore, the use of these terms in Footnotes (a) and (b) in ISTS Table 3.3.1-1 are revised in the ITS to include the plant specific terminology. To comply with the guidance provided in NRC Regulatory Issue Summary 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36, ‘Technical Specifications,’ Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels,” the PSL setpoint calculations have been structured to include a Nominal Trip Setpoint (NTSP), also known as the field trip setpoint, and determination of an OPERABILITY limit range, referred to in CTS as acceptance criteria band. For PSL, the Nominal Trip Setpoint (NTSP) is synonymous with the Limiting Trip Setpoint (LTSP) and the as-found acceptance criteria band (i.e., OPERABILITY limit range) is synonymous with the as-found tolerance.

5. This Reactor Coolant System Loops - Test Exceptions Specification is not included in the PSL Unit 1 and Unit 2 ITS because the exception is not needed to perform any required startup or PHYSICS TESTS.

6. The addition of Footnote (e) to Function 2 (Wide Range Logarithmic Neutron Flux Monitor Power Rate of Change – High) in ITS Table 3.3.1-1, reflects current licensing basis as approved in License Amendment 243 (Unit 1) and Amendment 194 (Unit 2), dated October 31, 2017 (NRC ADAMS Accession Number ML17257A015). In addition to allowing the Wide Range Logarithmic Neutron Flux Monitor Power Rate of Change – High RPS Function channels to be bypassed when THERMAL is > 15% RPT, this change allowed the RPS Function to not apply when THERMAL POWER is > 15% RTP. As stated in the NRC Safety Evaluation accompanying the license amendments, the NRC staff determined that the change to limit MODE 1 applicability for the Wide Range Logarithmic Neutron Flux Monitor Power Rate of Change – High RPS Function to ≤15% RTP neither physically changes any plant systems, structures, or components, nor modifies any plant procedure or methodology for this phase of plant operation, and therefore, is acceptable.

7. **Unit 2 only:** ISTS 3.3.1, Required Action A.2.2 is deleted and the Completion Time of Required Action A.2.1 (ITS 3.3.1, Required Action A.2) is modified in the Unit 2 ITS consistent with the PSL current licensing basis to allow the inoperable channel to be in trip or bypass until the next plant startup following entry into MODE 5. In

addition, ISTS Required Action B.2 is unnecessary since ACTION A requires restoration of the channel to OPERABLE status with no option to place the trip unit in the trip condition indefinitely. These changes to the ISTS 3.3.1 ACTIONS A and B are similar to ISTS 3.3.1 (Digital) ACTIONS A and B. The Unit 2 RPS design includes independence of the four protection channels to ensure no single failure (e.g., circuit fault) can adversely affect more than one channel. As discussed in Section 7.2.4 of NUREG-0843, "Safety Evaluation Report - St. Lucie Unit 2," the NRC concluded that the RPS design enhancements were sufficient to ensure the independence of the four protection channels. Thus, the Unit 2 Technical Specifications allow a failed channel to be placed in long term trip or bypass condition provided the channel is restored no later than the next cold shutdown following the channel malfunction. A Note is added to ITS 3.3.1 Condition A stating that Required Action A.2 shall be completed when Condition A is entered. This Note constitutes an "unless otherwise stated" exception to LCO 3.0.2 to require the action to be completed within the associated Completion Time even when the LCO is no longer applicable. The added Note is similar to other Notes in the ISTS (e.g., Note to Condition A of ISTS 3.2.3 and the Note to Conditions A and C of ISTS 3.4.3.).

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.1 Reactor Protective System (RPS) Instrumentation ~~– Operating (Analog)~~ 2

BASES

BACKGROUND

The Reactor Protective System (RPS) initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary (RCPB) during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulation as "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for channel uncertainties related to the setting at which the automatic protective action would actually occur.

~~---------- REVIEWER'S NOTE ----------~~

~~The term "Limiting Trip Setpoint" [LTSP] is generic terminology for the calculated trip setting (setpoint) value calculated by means of the plant specific setpoint methodology documented in a document controlled under 10 CFR 50.59. The term [LTSP] indicates that no additional margin has been added between the Analytical Limit and the calculated trip setting.~~

~~"Nominal Trip Setpoint [NTSP]" is the suggested terminology for the actual setpoint implemented in the plant surveillance procedures where margin has been added to the calculated [LTSP]. The as-found and as-left tolerances will apply to the [NTSP] implemented in the Surveillance procedures to confirm channel performance.~~ 4

BASES

BACKGROUND (continued)

~~Licensees are to insert the name of the document(s) controlled under 10 CFR 50.59 that contain the methodology for calculating the as-left and as-found tolerances in Note b of Table 3.3.1-1 for the phrase "[insert the name of a document controlled under 10 CFR 50.59 such as the Technical Requirements Manual or any document incorporated into the facility FSAR]" throughout these Bases.~~ 4

~~Where the [LTSP] is not included in Table 3.3.1-1, the plant specific location for the [LTSP] or [NTSP] must be cited in Note b of Table 3.3.1-1. The brackets indicate plant specific terms may apply, as reviewed and approved by the NRC.~~

Nominal NTSP Reference 1

The ~~[Limiting~~ Trip Setpoint (~~LTSP~~)~~]~~ specified in ~~Table 3.3.1-1~~ is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the ~~[LTSP]~~ accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the ~~[LTSP]~~ ensures that SLs are not exceeded. Therefore, the ~~[LTSP]~~ meets the definition of an LSSS (Ref. ~~1~~). 1 3 5 1

NTSP 2

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety function(s)." Relying solely on the ~~[LTSP]~~ to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a Surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the ~~[LTSP]~~ due to some drift of the setting may still be OPERABLE because drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the ~~[LTSP]~~ and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE because it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left ~~tolerance~~ around the ~~[LTSP]~~ to account for further drift during the next surveillance interval. Note that, 3

NTSP

acceptance criteria band NTSP 5

BASES

BACKGROUND (continued)

acceptance criteria band

although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left ~~tolerance~~ in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria). 5

However, there is also some point beyond which the channel may not be able to perform its function due to, for example, greater than expected drift. This value needs to be specified in the Technical Specifications in order to define OPERABILITY of the channels and is designated as the Allowable Value.

acceptance criteria band (e.g., conservative side of the as-found acceptance criteria band)

If the actual setting (as-found setpoint) of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found ~~tolerance band~~, the channel is OPERABLE, but degraded. The degraded condition will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the ~~[Nominal Trip Setpoint (~~NTSP~~)]~~ (within the allowed ~~tolerance~~), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation. 5 3

acceptance criteria band

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling,
- Fuel centerline melting shall not occur, and
- The Reactor Coolant System (RCS) pressure SL of ~~[~~2750~~]~~ psia shall not be exceeded. 3

Maintaining the parameters within the above values ensures that the offsite dose will be within the ~~10 CFR 50 (Ref. 2) and~~ 10 CFR ~~100~~ (Ref. ~~3~~) criteria during AOOs. 1

4

50.67

BASES

BACKGROUND (continued)

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~ (Ref. ~~3~~) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event. 1

50.67 4

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels,
- Bistable trip units,
- RPS Logic, and
- Reactor trip circuit breakers (RTCBs).

This LCO addresses measurement channels and bistable trip units. It also addresses the automatic bypass removal feature for those trips with operating bypasses. The RPS Logic and RTCBs are addressed in LCO 3.3.~~3~~, "Reactor Protective System (RPS) Logic and Trip Initiation." 1

2

The role of each of these modules in the RPS, including those associated with the logic and RTCBs, is discussed below.

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

The excore nuclear instrumentation and the analog core protection calculators (CPCs) are considered components in the measurement channels. The wide range nuclear instruments (NIs) provide a Power Rate of Change - High Trip. Three RPS trips use a power level designated as Q power as an input. Q power is the higher of NI power and primary calorimetric power (ΔT power) based on RCS hot leg and cold leg temperatures. Trips using Q power as an input include the Variable ~~High~~ Power Trip (VHPT) - High, Thermal Margin/Low Pressure (TM/LP), and the ~~Axial~~ Power ~~Distribution (APD)~~ - High trips. 1

Level Local Density

BASES

BACKGROUND (continued)

Local Power Density

The analog CPCs provide the complex signal processing necessary to calculate the TM/LP trip setpoint, ~~APD~~ trip setpoint, VHPT trip setpoint, and Q power calculation. 1

linear

Local Power Density

The excore NIs (wide range and ~~power~~ range) and the analog CPCs (TM/LP and ~~APD~~ calculators) are mounted in the RPS cabinet, with one channel of each in each of the four RPS bays. 1

Four identical measurement channels, designated channels A through D, with electrical and physical separation are provided for each parameter used in the direct generation of trip signals. Measurement channels provide input to one or more RPS bistables within the same RPS channel. In addition, some measurement channels may also be used as inputs to Engineered Safety Features Actuation System (ESFAS) bistables, and most provide indication in the control room. Measurement channels used as an input to the RPS are never used for control functions.

When a channel monitoring a parameter exceeds a predetermined setpoint, indicating an unsafe condition, the bistable monitoring the parameter in that channel will trip. Tripping two or more channels of

the

bistables monitoring the same parameter de-energizes Matrix Logic, 7
which in turn de-energizes the Initiation Logic. This causes all eight RTCBs to open, interrupting power to the control element assemblies (CEAs), allowing them to fall into the core.

Three of the four measurement and bistable channels are necessary to meet the redundancy and testability of GDC 21 in 10 CFR 50, Appendix A (Ref. ~~2~~ 3). The fourth channel provides additional flexibility by allowing one 1
channel to be removed from service (trip channel bypass) for maintenance or testing while still maintaining a minimum two-out-of-three logic. Thus, even with a channel inoperable, no single additional failure in the RPS can either cause an inadvertent trip or prevent a required trip from occurring.

Since no single failure will either cause or prevent a protective system actuation, and no protective channel feeds a control channel, this arrangement meets the requirements of IEEE Standard 279-1971 (Ref. ~~4~~ 5). 1

NTSP

Many of the RPS trips are generated by comparing a single measurement to a fixed bistable ~~[LTSP]~~. Certain Functions, however, make use of more 3
than one measurement to provide a trip. The following trips use multiple measurement channel inputs:

BASES

BACKGROUND (continued)

- Steam Generator Level - Low

  This trip uses the lower of the two steam generator levels as an input to a common bistable.

- Steam Generator Pressure - Low

  This trip uses the lower of the two steam generator pressures as an input to a common bistable.

- Variable ~~High~~ Power Trip (VHPT) - High (Level) (1)

  The VHPT uses Q power as its only input. Q power is the higher of NI power and ΔT power. It has a trip setpoint that tracks power levels downward so that it is always within a fixed increment above current power, subject to a minimum value.

  On power increases, the trip setpoint remains fixed unless manually reset, at which point it increases to the new setpoint, a fixed increment above Q power at the time of reset, subject to a maximum value. Thus, during power escalation, the trip setpoint must be repeatedly reset to avoid a reactor trip.

- Thermal Margin/Low Pressure (TM/LP) and Steam Generator Pressure Difference

  Q power is only one of several inputs to the TM/LP trip. Other inputs include internal ASI and cold leg temperature based on the higher of two cold leg resistance temperature detectors. The TM/LP trip setpoint is a complex function of these inputs and represents a minimum acceptable RCS pressure to be compared to actual RCS pressure in the TM/LP trip unit.

  Steam generator pressure is also an indirect input to the TM/LP trip via the Steam Generator Pressure Difference. This Function provides a reactor trip when the secondary pressure in either steam generator exceeds that of the other generator by greater than a fixed amount. The trip is implemented by biasing the TM/LP trip setpoint upward so as to ensure TM/LP trip if an asymmetric steam generator transient is detected.

- ~~Axial Power Distribution (APD)~~ – High

  Q Power and ASI are inputs to the ~~APD~~ trip. The ~~APD~~ trip setpoint is a function of Q power, being more restrictive at higher power levels. It provides a reactor trip if actual ASI exceeds the ~~APD~~ trip setpoint. (Local Power Density) (Local Power Density) (1)

BASES

BACKGROUND (continued)

Bistable Trip Units

Bistable trip units, mounted in the RPS cabinet, receive an analog input from the measurement channels, compare the analog input to trip setpoints, and provide contact output to the Matrix Logic. They also provide local trip indication and remote annunciation.

There are four channels of bistable trip units, designated A through D, for each RPS Function, one for each measurement channel. Bistable output relays de-energize when a trip occurs.

The contacts from these bistable relays are arranged into six coincidence matrices, comprising the Matrix Logic. If bistables monitoring the same parameter in at least two channels trip, the Matrix Logic will generate a reactor trip (two-out-of-four logic).

Some of the RPS measurement channels provide contact outputs to the RPS, so the comparison of an analog input to a trip setpoint is not necessary. In these cases, the bistable trip unit is replaced with an auxiliary trip unit. The auxiliary trip units provide contact multiplication so the single input contact opening can provide multiple contact outputs to the coincidence logic as well as trip indication and annunciation.

Local Power Density

turbine generator control system hydraulic

Local Power Density

Trips employing auxiliary trip units include the Loss of Load trip and the ~~APD~~ - High trip. The Loss of Load trip is a contact input from the ~~Electro Hydraulic Control System~~ control oil pressure ~~on each of the~~ four high pressure ~~stop valves~~.

switches mounted on the emergency trip fluid line common header.

sensed by

1

R1

The ~~APD~~ trip, described above, is a complex function in which the actual trip comparison is performed within the CPC. Therefore the ~~APD~~ - High trip unit employs a contact input from the CPC.

1

Local Power Density

All RPS trips, with the exception of the Loss of Load trip, generate a pretrip alarm as the trip setpoint is approached.

6

The trip setpoints used in the bistable trip units are based on the analytical limits stated in Reference ~~5~~. The calculation of the ~~[LTSP]~~ specified in Table 3.3.1-1 is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration ~~tolerances~~, instrumentation uncertainties, instrument drift, and severe environment errors - for those RPS channels that must function in harsh environments, as defined by 10 CFR 50.49 (Ref. ~~6~~) - Allowable Values specified in Table 3.3.1-1, in the accompanying LCO,

NTSP

3

acceptance criteria bands

5

7

1

BASES

BACKGROUND (continued)

are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in ~~the "Plant Protection System Selection of Trip Setpoint Values"~~ (Ref. ~~7~~). The trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value, to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the interval between surveillances.

IC-3.17, "Instrument Setpoint Methodology for Nuclear Power Plants" 8 1

The ~~[~~LTSP~~]~~ is the value at which the bistable is set and is the expected value to be achieved during calibration. The ~~[~~LTSP~~]~~ value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties.

NTSP 3

~~[~~LTSPs~~]~~, in conjunction with the use of as-found and as-left ~~tolerances~~, consistent with the requirements of the Allowable Value will ensure that SLs of Chapter 2.0 are not violated during AOOs and the consequences of DBAs will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

acceptance criteria bands 3 5

Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the least conservative value of the as-found setpoint that a channel can have during a periodic CHANNEL CALIBRATION or CHANNEL FUNCTIONAL TEST.

RPS Logic

The RPS Logic, addressed in LCO 3.3.~~3~~, consists of both Matrix and Initiation Logic and employs a scheme that provides a reactor trip when bistables in any two out of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic. ~~This logic and the RTCB configuration are shown in Figure B 3.3.1-1.~~

2 1 6

Bistable relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

BASES

BACKGROUND (continued)

The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized RTCB control relays (K1, K2, K3, and K4). The trip paths thus each have six contacts in series, one from each matrix, and perform a logical OR function, opening the RTCBs if any one or more of the six logic matrices indicate a coincidence condition.

Each trip path is responsible for opening one set of two of the eight RTCBs. The RTCB control relays (K-relays), when de-energized, interrupt power to the breaker undervoltage trip attachments and simultaneously apply power to the shunt trip attachments on each of the two breakers. Actuation of either the undervoltage or shunt trip attachment is sufficient to open the RTCB and interrupt power from the motor generator (MG) sets to the control element drive mechanisms (CEDMs).

When a coincidence occurs in two RPS channels, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four RTCB control relays, which simultaneously de-energize the undervoltage and energize the shunt trip attachments in all eight RTCBs, tripping them open.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting matrix wiring between bistable and auxiliary trip units, up to but not including the matrix relays. Contacts in the bistable and auxiliary trip units are excluded from the Matrix Logic definition, since they are addressed as part of the measurement channel.

The Initiation Logic consists of the trip path power source, matrix relays and their associated contacts, all interconnecting wiring, and solid state (auxiliary) relays through the K-relay contacts in the RTCB control circuitry.

It is possible to change the two-out-of-four RPS Logic to a two-out-of-three logic for a given input parameter in one channel at a time by trip channel bypassing select portions of the Matrix Logic. Trip channel bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional channels indicate a trip condition. Trip channel bypassing can be simultaneously performed on

BASES

BACKGROUND (continued)

A mechanical interlock (key capture) in conjunction with administrative control

any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. ~~An interlock~~ prevents simultaneous trip channel bypassing of the same parameter in more than one channel. Trip channel bypassing is normally employed during maintenance or testing. 1

For those plants that have demonstrated sufficient channel to channel independence, two-out-of-three logic is the minimum that is required to provide adequate plant protection, since a failure of one channel still ensures a reactor trip would be generated by the two remaining OPERABLE channels. Two-out-of-three logic also prevents inadvertent trips caused by any single channel failure in a trip condition.

Zero Power Mode trips (Reactor Coolant Flow – Low and Thermal Margin/Low Pressure), Local Power Density - High, Loss of Load -High,

Local Power Density

In addition to the trip channel bypasses, there are also operating bypasses on select RPS trips. Some of these bypasses are enabled manually, others automatically, in all four RPS channels when plant conditions do not warrant the specific trip protection. All operating bypasses are automatically removed when enabling bypass conditions are no longer satisfied. Trips with operating bypasses include Power Rate of Change - High, ~~Reactor Coolant Flow - Low,~~ Steam Generator Pressure - Low, ~~APD - High, TM/LP,~~ and Steam Generator Pressure Difference. ~~[~~The Loss of Load trip, Power Rate of Change - High, and ~~APD~~ - High operating bypasses are automatically enabled and disabled.~~]~~ 1 3 R1

The Variable Power Level Trip – High bypass is manual enable but automatically

Reactor Trip Circuit Breakers (RTCBs)

2

The reactor trip switchgear, addressed in LCO 3.3.~~3 and shown in Figure B 3.3.1-1~~, consists of eight RTCBs, which are operated in four sets of two breakers (four channels). Power input to the reactor trip switchgear comes from two full capacity MG sets operated in parallel such that the loss of either MG set does not de-energize the CEDMs. There are two separate CEDM power supply buses, each bus powering half of the CEDMs. Power is supplied from the MG sets to each bus via two redundant paths (trip legs). Trip legs 1A and 1B supply power to CEDM bus 1. Trip legs 2A and 2B supply power to CEDM bus 2. This ensures that a fault or the opening of a breaker in one trip leg (i.e., for testing purposes) will not interrupt power to the CEDM buses. 6

Each of the four trip legs consists of two RTCBs in series. The two RTCBs within a trip leg are actuated by separate initiation circuits.

BASES

BACKGROUND (continued)

The eight RTCBs are operated as four sets of two breakers (four channels). For example, if a breaker receives an open signal in trip leg A (for CEDM bus 1), an identical breaker in trip leg B (for CEDM bus 2) will also receive an open signal. This arrangement ensures that power is interrupted to both CEDM buses, thus preventing trip of only half of the CEAs (a half trip). Any one inoperable breaker in a channel will make the entire channel inoperable.

Each set of RTCBs is operated by either a Manual Trip push button or an RPS actuated K-relay. There are four Manual Trip push buttons, arranged in two sets of two~~, as shown in Figure B 3.3.1-1~~. Depressing both push buttons in either set will result in a reactor trip. 6

When a Manual Trip is initiated using the control room push buttons, the RPS trip paths and K-relays are bypassed, and the RTCB undervoltage and shunt trip attachments are actuated independent of the RPS.

Manual Trip circuitry includes the push button and interconnecting wiring to both RTCBs necessary to actuate both the undervoltage and shunt trip attachments but excludes the K-relay contacts and their interconnecting wiring to the RTCBs, which are considered part of the Initiation Logic.

Functional testing of the entire RPS, from bistable input through the opening of individual sets of RTCBs, can be performed either at power or shutdown and is ~~normally~~ performed ~~on a quarterly basis~~. per the Surveillance Frequency Control Program 1 3
FSAR, Section ~~[~~7.2~~]~~ (Ref. ~~3~~), explains RPS testing in more detail. U 9

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Reference ~~5~~ takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis are part of the NRC approved licensing basis for the plant. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. Other Functions, such as the Loss of Load trip, are purely equipment protective, and their use minimizes the potential for equipment damage. 6 1

Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The specific safety analyses applicable to each protective Function are identified below:

1. Variable ~~High~~ Power Trip (VHPT) - High

Level

The VHPT provides reactor core protection against positive reactivity excursions that are too rapid for a Pressurizer Pressure - High or TM/LP trip to protect against. The following events require VHPT protection:

- Uncontrolled CEA withdrawal event,
- Excess load,
- ~~Excess feedwater heat removal event,~~
- CEA ejection event, and
-
- Main steam line break (MSLB) ~~(outside containment)~~.

1

R1

two

third and fourth

The first t~~hree~~ events are AOOs, and fuel integrity is maintained. The ~~fourth and fifth~~ are accidents, and limited fuel damage may occur.

2. Power Rate of Change - High

The Power Rate of Change - High trip is used to trip the reactor when excore ~~[~~logarithmic~~]~~ power indicates an excessive rate of change. The Power Rate of Change - High Function ~~minimizes transients for events such as a continuous CEA withdrawal or a boron dilution event from low power levels~~. The trip may be bypassed when THERMAL POWER is < 1E-4% RTP, when poor counting statistics may lead to erroneous indication. ~~It is also~~ bypassed at > 15% RTP, where moderator temperature coefficient and fuel temperature coefficient make high rate of change of power unlikely. With the RTCBs open, the Power Rate of Change - High trip is not required to be OPERABLE; however, the indication ~~and alarm~~ Function~~s~~ of at least two channels ~~are~~ required by LCO 3.3.1~~3~~, "~~[~~Logarithmic~~]~~ ~~Power~~ Monitoring ~~Channels~~," to be OPERABLE. LCO 3.3.1~~3~~ ensures the ~~[~~logarithmic~~]~~ channels are available to detect ~~and alert the operator to~~ a boron dilution event.

is not credited in the accident analysis as providing primary protection for any limiting case AOO or accident.

3
1

In addition, the trip is not required to be OPERABLE and is

Neutron Flux

is

3
1
1
3
1
3
1

BASES

APPLICABLE SAFETY ANALYSES (continued)

3. Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection during the following events:

- Loss of RCS flow,
- Loss of ~~nonemergency~~ AC power, and
- Reactor coolant pump (RCP) seized /sheared shaft~~,~~
- ~~RCP sheared shaft, and~~
- ~~Certain MSLB events~~.

The loss of RCS flow and ~~of nonemergency~~ AC power events are AOOs where fuel integrity is maintained. The RCP seized shaft~~,~~ and sheared shaft~~, and MSLBs~~ are accidents where fuel damage may result.

4. Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with pressurizer safety valves and main steam safety valves (MSSVs), provides protection against overpressure conditions in the RCS during the following events:

- Loss of condenser vacuum with a concurrent loss of offsite power,
- Loss of condenser vacuum with a concurrent loss of one 6.9 kV bus,
- ~~Isolation of turbine at 102% power~~, Loss of External Load/Turbine trip
- ~~Feedwater System pipe breaks between the steam generator and check valve~~,
- CEA withdrawal, and
- Loss of feedwater flow.

1

R1

BASES

APPLICABLE SAFETY ANALYSES (continued)

5. Containment Pressure - High

The Containment Pressure - High trip prevents exceeding the containment design pressure during certain loss of coolant accidents (LOCAs) or feedwater line break accidents. It ensures a reactor trip prior to, or concurrent with, a LOCA, thus assisting the ESFAS in the event of a LOCA or MSLB. Since these are accidents, SLs may be violated. However, the consequences of the accident will be acceptable.

6. Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection against an excessive rate of heat extraction from the steam generators, which would result in a rapid uncontrolled cooldown of the RCS. This trip is needed to shut down the reactor and assist the ESFAS in the event of an MSLB. Since these are accidents, SLs may be violated. However, the consequences of the accident will be acceptable.

7.a, 7.b. Steam Generator A and B Level - Low

The Steam Generator A Level - Low and Steam Generator B Level - Low trips are required for the following events:

- ~~Steam System piping failures,~~
- Feedwater System pipe breaks,
- ~~Inadvertent opening of a steam generator atmospheric dump valve (ADV),~~
- Loss of normal feedwater, and
- Asymmetric loss of feedwater.

Excess Load

1
R1

The Steam Generator Level - Low trip ensures that low DNBR, high local power density, and the RCS pressure SLs are maintained during normal operation and AOOs, and, in conjunction with the ESFAS, the consequences of the Feedwater System pipe break accident will be acceptable.

BASES

APPLICABLE SAFETY ANALYSES (continued)

8. ~~Axial Power Distribution (APD)~~ - High

Local Power Density

The ~~APD~~ - High trip ensures that excessive axial peaking, such as that due to axial xenon oscillations, will not cause fuel damage. It ensures that neither a DNBR less than the SL nor a peak linear heat rate that corresponds to the temperature for fuel centerline melting will occur. This trip is the primary protection against fuel centerline melting. 1

9. Thermal Margin

a. Thermal Margin/Low Pressure (TM/LP)

The TM/LP trip prevents exceeding the DNBR SL during AOOs and aids the ESFAS during certain accidents. The following events require TM/LP protection:

- Excess load ~~(inadvertent opening of a steam generator ADV)~~, R1
- RCS depressurization (inadvertent safety or power operated relief valves (PORVs) opening), R1
- Steam generator tube rupture, and
- LOCA accident.

The first two events are AOOs, and fuel integrity is maintained. The third and fourth are accidents, and limited fuel damage may occur although only the LOCA is expected to result in fuel damage. The trip is initiated whenever the RCS pressure signal drops below a minimum value (P_{min}) or a computed value (P_{var}) as described below, whichever is higher. The computed value is a Function Q power, ASI, as determined from the axially split excore detectors~~,~~ and reactor inlet (cold leg) temperature~~, and the number of RCPs operating~~. 1

and

The minimum value of reactor coolant flow rate, the maximum ~~T~~$_Q$, and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip Function. In addition, CEA group sequencing in accordance with LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits," is assumed. Finally, the maximum insertion of CEA banks that can occur during any AOO prior to a VHPT is assumed. 1

azimuthal tilt

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. Steam Generator Pressure Difference

The Steam Generator Pressure Difference provides protection for those AOOs associated with secondary system malfunctions that result in asymmetric primary coolant temperatures. The most limiting event is closure of a single main steam isolation valve. Steam Generator Pressure Difference is provided by comparing the secondary pressure in both steam generators in the TM/LP calculator. If the pressure in either exceeds that in the other by the trip setpoint, a TM/LP trip will result.

10. Loss of Load

The Loss of Load ~~(~~turbine ~~stop valve (TSV) control oil pressure~~) trip is anticipatory for the loss of heat removal capabilities of the secondary system following a turbine trip. The Loss of Load trip prevents lifting the pressurizer safety valves, PORVs, and MSSVs in the event of a turbine generator trip. Thus, the trip minimizes the pressure and temperature transients on the reactor by initiating a trip well before reaching the Pressurizer Pressure - High trip and pressurizer safety valve setpoints. The four RPS Loss of Load reactor trip channels receive their input from ~~sensors mounted on the high pressure TSV actuators. Since there are four high pressure TSVs, one actuator per valve and one sensor per actuator, each sensor sends its signal to a different RPS channel. When the turbine trips, control oil is dumped from the high pressure TSVs. When the control oil pressure drops to the appropriate setpoint, a reactor trip signal is generated.~~

Hydraulic Fluid Pressure - Low

four redundant pressure switches mounted on the emergency trip fluid line common header, to serve as the loss of load, turbine trip input to the Reactor Protective System logic matrices. Actuation of any two of the pressure switches on low hydraulic oil pressure causes a reactor trip.

R1 1

1

Interlocks/Bypasses

The bypasses and their Allowable Values are addressed in footnotes to Table 3.3.1-1. They are not otherwise addressed as specific Table entries.

The automatic bypass removal features must function as a backup to manual actions for all safety related trips to ensure the trip Functions are not operationally bypassed when the safety analysis assumes the Functions are not bypassed. The RPS operating bypasses are:

Zero power mode bypass (ZPMB) removal on the TM/LP, Steam Generator Pressure Difference, and reactor coolant low flow trips when THERMAL POWER is < 1~~E-4~~% RTP. This bypass is manually enabled below the specified setpoint to permit low power testing. The wide range NI Level 1 bistable in the wide range drawer permits manual bypassing below the setpoint and removes the bypass above the setpoint.

1 R1

BASES

APPLICABLE SAFETY ANALYSES (continued)

Power rate of change bypass removal. The Power Rate of Change - High trip is automatically bypassed at < 1E-4% RTP, as sensed by the wide range NI Level 2 bistable, and at > ~~12~~% RTP by the power range NI Level 1 bistable, mounted in their respective NI drawers. Automatic bypass removal is also effected by these bistables when conditions are no longer satisfied.

Local Power Density

Loss of Load and ~~APD~~ - High bypass removal. The Loss of Load and ~~APD~~ - High trips are automatically bypassed when at < 15% RTP as sensed by the power range NI Level 1 bistable. The bypass is automatically removed by this bistable above the setpoint. This same bistable is used to bypass the Power Rate of Change - High trip.

Steam Generator Pressure - Low bypass removal. The Steam Generator Pressure - Low trip is manually enabled below the pretrip setpoint. The permissive is removed, and the bypass automatically removed, when the Steam Generator Pressure - Low pretrip clears.

The RPS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The specific criteria for determining channel OPERABILITY differ slightly between Functions. These criteria are discussed on a Function by Function basis below.

Actions allow maintenance (trip channel) bypass of individual channels, but ~~the bypass activates interlocks that prevent~~ operation with a second channel in the same Function bypassed. Plants are restricted to 48 hours in a trip channel bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic). ~~At plants where adequate channel to channel independence has been demonstrated, specific exceptions may be approved by the NRC staff to permit one of the two-out-of-four channels to be bypassed for an extended period of time.~~

is prevented by mechanical interlock (key capture)

the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively

Allowable Values for RPS Instrumentation Functions are specified in Table 3.3.1-1. ~~[~~LTSPs~~]~~ and the methodologies for calculation of the as-left and as-found ~~tolerances~~ are described in ~~[insert the name of a document controlled under 10 CFR 50.59 such as the Technical Requirements Manual or any document incorporated into the facility FSAR]~~. The ~~[~~LTSPs~~]~~ are selected to ensure that the actual setpoints remain conservative with respect to the as-found ~~tolerance~~ band between successive CHANNEL CALIBRATIONS. After each calibration the trip setpoint shall be left within the as-left band around the ~~[~~LTSP~~]~~.

NTSPs

NTSPs

acceptance criteria

NTSP

BASES

LCO (continued)

The following Bases for each trip Function identify the above RPS trip Function criteria items that are applicable to establish the trip Function OPERABILITY.

Level

1. Variable ~~High~~ Power Trip (VHPT) - High 1

This LCO requires all four channels of the VHPT to be OPERABLE in MODES 1 and 2.

The Allowable Value is high enough to provide an operating envelope that prevents unnecessary Linear Power Level - High reactor VHPT - High trips during normal plant operations. The Allowable Value is low enough for the system to maintain a margin to unacceptable fuel cladding damage should a CEA ejection accident occur.

The VHPT setpoint is operator adjustable and can be set at a fixed increment above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The ~~[LTSP]~~ has a maximum and a minimum setpoint. 3

NTSP

NTSP

Adding to this maximum value the possible variation in ~~[LTSP]~~ due to calibration and instrument errors, the maximum actual steady state THERMAL POWER level at which a trip would be actuated is ~~112~~% RTP, which is the value used in the safety analyses. 3

107

R1

conservative with respect to

of 112% RTP

1

Calibration and instrument errors are considered within the setpoint calculations. The stated minimum and maximum Allowable Values and step values are the assumed safety analysis values.

~~To account for these errors, the safety analysis minimum value is 40% RTP. The 10% step is a maximum value assumed in the safety analysis. There is no uncertainty applied to the step.~~

R1

2. Power Rate of Change - High

MODE 1 with THERMAL POWER ≤ 15% RTP and MODE 2

This LCO requires four channels of Power Rate of Change - High to be OPERABLE in ~~MODES 1 and 2, as well as in MODES 3, 4, and 5 when the RTCBs are closed and the CEA Drive System is capable of CEA withdrawal~~. 1

The high power rate of change trip serves as a backup to the administratively enforced startup rate limit. The Function is not credited in the accident analyses; therefore, the Allowable Value for the trip or bypass Functions is not derived from analytical limits.

BASES

LCO (continued)

3. Reactor Coolant Flow - Low

This LCO requires four channels of Reactor Coolant Flow - Low to be OPERABLE in MODES 1 and 2.

1 The trip may be manually bypassed when THERMAL POWER falls below ~~1E-4~~% RTP. This bypass is part of the ZPMB circuitry, which also bypasses the TM/LP trip and provides a ΔT power block signal to the Q power select logic. This ZPMB allows low power physics testing at reduced RCS temperatures and pressures. It also allows heatup and cooldown with shutdown CEAs withdrawn.

This trip is set high enough to maintain fuel integrity during a loss of flow condition. The setting is low enough to allow for normal operating fluctuations from offsite power. ~~To account for analysis uncertainty, the value in the safety analysis is 93% RTP.~~

4. Pressurizer Pressure - High

This LCO requires four channels of Pressurizer Pressure - High to be OPERABLE in MODES 1 and 2.

The Allowable Value is set high enough to allow for pressure increases in the RCS during normal operation (i.e., plant transients) not indicative of an abnormal condition. The setting is below the lift setpoint of the pressurizer safety valves and low enough to initiate a reactor trip when an abnormal condition is indicated. ~~The difference between the Allowable Value and the analysis setpoint of 2470 psia includes allowance for harsh environment.~~

The Pressurizer Pressure - High trip concurrent with PORV operation avoids unnecessary operation of the pressurizer safety valves.

5. Containment Pressure - High

This LCO requires four channels of Containment Pressure - High to be OPERABLE in MODES 1 and 2.

The Allowable Value is high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) that are not indicative of an abnormal condition. The setting is low enough to initiate a reactor trip to prevent containment pressure from exceeding design pressure following a DBA.

BASES

LCO (continued)

6. Steam Generator Pressure - Low

This LCO requires four channels of Steam Generator Pressure - Low per steam generator to be OPERABLE in MODES 1 and 2.

The Allowable Value is sufficiently below the full load operating value for steam pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of excessive steam demand. Since excessive steam demand causes the RCS to cool down, resulting in positive reactivity addition to the core, a reactor trip is required to offset that effect.

~~The difference between the Allowable Value and the safety analysis value of 600 psia includes harsh environment uncertainties.~~ 1

The Function may be manually bypassed as steam generator pressure is reduced during controlled plant shutdowns. This bypass is permitted at a preset steam generator pressure. The bypass, in conjunction with the ZPMB, allows testing at low temperatures and pressures, and heatup and cooldown with the shutdown CEAs withdrawn. From a bypass condition the trip will be reinstated automatically as steam generator pressure increases above the preset pressure.

7.a, 7.b. Steam Generator Level – Low

This LCO requires four channels of Steam Generator Level - Low per steam generator to be OPERABLE in MODES 1 and 2.

The Allowable Value is sufficiently below the normal operating level for the steam generators so as not to cause a reactor trip during normal plant operations. The trip setpoint is high enough to ensure a reactor trip signal is generated before water level drops below the top of the feed ring~~. The difference between the Allowable Value and the measurement value includes 10 inches of measurement uncertainty. The specified setpoint ensures there will be sufficient water inventory to provide a 10 minute margin before auxiliary feedwater is required for the removal of decay heat.~~ 1

BASES

LCO (continued)

8. ~~Axial Power Distribution (APD)~~ - High

Local Power Density

This LCO requires four channels of ~~APD~~ - High to be OPERABLE in MODE 1 ≥ 15% RTP.

The Allowable Value curve was derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore ASI relationship. 1

Local Power Density

The ~~APD~~ trip is automatically bypassed at < 15% RTP, where it is not required for reactor protection.

9. Thermal Margin

a. Thermal Margin/Low Pressure (TM/LP)

This LCO requires four channels of TM/LP to be OPERABLE in MODES 1 and 2.

The Allowable Value includes allowances for equipment response time, measurement uncertainties, processing error, and a further allowance to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the SL.

1

This trip may be manually bypassed when THERMAL POWER falls below ~~1E-4~~% RTP. This bypass is part of the ZPMB circuitry, which also bypasses the Reactor Coolant Flow - Low trip and provides a ΔT power block signal to the Q power select logic. This ZPMB allows low power physics testing at reduced RCS temperatures and pressures. It also allows heatup and cooldown with shutdown CEAs withdrawn. 1

b. Steam Generator Pressure Difference

This LCO requires four channels of Steam Generator Pressure Difference to be OPERABLE in MODES 1 and 2.

The Allowable Value is high enough to avoid trips caused by normal operation and minor transients, but ensures DNBR protection in the event of Design Basis Events. ~~The difference between the Allowable Value and the 175 psia analysis setpoint allows for 40 psia of measurement uncertainty.~~ 1

BASES

LCO (continued)

1 The trip may be bypassed when THERMAL POWER falls below ~~1E-4~~% RTP. The Steam Generator Pressure Difference is subject to the ZPMB, since it is an input to the TM/LP trip and is not required for protection at low power levels. 1

10. Loss of Load

The LCO requires four Loss of Load trip channels to be OPERABLE in MODE 1 ≥ 15% RTP.

The Loss of Load trip may be bypassed when THERMAL POWER falls below 15%, since it is no longer needed to prevent lifting of the pressurizer safety valves, steam generator safety valves, or PORVs in the event of a Loss of Load. The Nuclear Steam Supply System and the Steam Dump System are capable of accommodating the Loss of Load without requiring the use of the above equipment.

Interlocks/Bypasses

The LCO on bypass permissive removal channels requires that the automatic bypass removal feature of all four operating bypass channels be OPERABLE for each RPS Function with an operating bypass in the MODES addressed in the specific LCO for each Function. All four bypass removal channels must be OPERABLE to ensure that none of the four RPS channels are inadvertently bypassed.

The LCO applies to the bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue.

The interlock Allowable Values are based on analysis requirements for the bypassed functions. These are discussed above as part of the LCO discussion for the affected Functions.

APPLICABILITY This LCO is applicable in accordance with Table 3.3.1-1. Most RPS trips are required to be OPERABLE in MODES 1 and 2 because the reactor is critical in these MODES. The trips are designed to take the reactor subcritical, maintaining the SLs during AOOs and assisting the ESFAS in providing acceptable consequences during accidents. Exceptions are addressed in footnotes to the table. Exceptions to this APPLICABILITY are:

Local Power Density

- The ~~APD~~ - High Trip and Loss of Load are only applicable in MODE 1 ≥ 15% RTP because they may be automatically bypassed at < 15% RTP, where they are no longer needed. 1

BASES

APPLICABILITY (continued)

- ~~The Power Rate of Change - High trip, RPS Logic, RTCBs, and Manual Trip are also required in MODES 3, 4, and 5, with the RTCBs closed, to provide protection for boron dilution and CEA withdrawal events. The Power Rate of Change - High trip in these lower MODES is addressed in LCO 3.3.2, "Reactor Protective System (RPS) Instrumentation - Shutdown." The RPS Logic in MODES 1, 2, 3, 4, and 5 is addressed in LCO 3.3.3.~~

1

The RPS

~~Most~~ trips are not required to be OPERABLE in MODES 3, 4, and 5. In MODES 3, 4, and 5, the emphasis is placed on return to power events. The reactor is protected in these MODES by ensuring adequate SDM.

ACTIONS

acceptance criteria band

The most common causes of channel inoperability are outright failure or drift of the bistable or process module sufficient to exceed the ~~tolerance~~ allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the trip setpoint is nonconservative with respect to the Allowable Value in Table 3.3.1-1, the channel is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.

5

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the channel is not functioning as required, or the transmitter, instrument loop, signal processing electronics, or RPS bistable trip unit is found inoperable, then all affected Functions provided by that channel must be declared inoperable, and the plant must enter the Condition for the particular protection Function affected.

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Times of each inoperable Function will be tracked separately for each Function, starting from the time the Condition was entered.

BASES

ACTIONS (continued)

A.1, A.2.1, and A.2.2

Condition A applies to the failure of a single channel in any RPS automatic trip Function. RPS coincidence logic is normally two-out-of-four.

If one RPS bistable trip unit or associated instrument channel is inoperable, startup or power operation is allowed to continue, providing the inoperable trip unit is placed in bypass or trip within 1 hour (Required Action A.1). With one channel in bypass, no additional random failure of a single channel could spuriously trip the reactor and a valid trip signal can still trip the reactor. With one channel in trip, an additional random failure of a single channel could spuriously trip the reactor. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to restore, bypass, or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel while ensuring that the risk involved in operating with the failed channel is acceptable.

The failed channel is restored to OPERABLE status or is placed in trip within ~~[~~48~~]~~ hours ~~[or in accordance with the Risk Informed Completion Time Program]~~ (Required Action A.2.1 or Required Action A.2.2). Required Action A.2.1 restores the full capability of the Function.

~~[~~Required Action A.2.2 places the Function in a one-out-of-three configuration. In this configuration, common cause failure of dependent channels cannot prevent trip.~~]~~

The Completion Time of ~~[~~48~~]~~ hours is based on operating experience, which has demonstrated that a random failure of a second channel occurring during the ~~[~~48~~]~~ hour period is a low probability event. 3

B.1 and B.2

Condition B applies to the failure of two channels in any RPS automatic trip Function.

Required Action B.1 provides for placing one inoperable channel in bypass and the other channel in trip within the Completion Time of 1 hour. This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels while ensuring that the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed, the RPS is in a two-out-of-three logic; but with another channel failed, the RPS may be operating

BASES

ACTIONS (continued)

in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

One channel should be restored to OPERABLE status within ~~[~~48~~]~~ hours ~~[or in accordance with the Risk Informed Completion Time Program]~~ for reasons similar to those stated under Condition A. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 must be placed in trip if more than ~~[~~48~~]~~ hours have elapsed since the initial channel failure. 3

C.1 and C.2

Local Power Density

The excore detectors are used to generate the internal ASI used as an input to the TM/LP and ~~APD~~ - High trips. Incore detectors provide a more accurate measurement of ASI. If one or more excore detectors cannot be calibrated to match incore detectors, power is restricted or reduced during subsequent operations because of increased uncertainty associated with using uncalibrated excore detectors. 3

The Completion Time of 24 hours is adequate to perform the SR while minimizing the risk of operating in an unsafe condition.

D.1, D.2.1, D.2.2.1, and D.2.2.2

Condition D applies to one automatic bypass removal channel inoperable. If the bypass removal channel for any operating bypass cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected RPS channel must be declared inoperable, as in Condition A, and the bypass either removed or the bypass removal channel repaired. The Bases for Required Actions and Completion Times are the same as discussed for Condition A.

E.1, E.2.1, and E.2.2

Condition E applies to two inoperable automatic bypass removal channels. If the bypass removal channels cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected

BASES

ACTIONS (continued)

RPS channels must be declared inoperable, as in Condition B, and the bypass either removed or the bypass removal channel repaired. Also, Required Action E.2.2 provides for the restoration of the one affected automatic trip channel to OPERABLE status within the rules of Completion Time specified under Condition B. Completion Times are consistent with Condition B.

F.1

Condition F is entered when the Required Action and associated Completion Time of Conditions A, B, C, D, or E are not met for the ~~Axial Power Distribution~~ Local Power Density and Loss of Load Trip Functions. 1

If the Required Actions associated with these Conditions cannot be completed within the required Completion Times, the reactor must be brought to a MODE in which the Required Actions do not apply. The allowed Completion Time of 6 hours to reduce THERMAL POWER to < 15% RTP is reasonable, based on operating experience, to decrease power to < 15% RTP from full power conditions in an orderly manner and without challenging plant systems.

G.1

Condition G is entered when the Required Action and associated Completion Time of Conditions A, B, C, D, E, or F are not met.

If the Required Actions associated with these Conditions cannot be completed within the required Completion Times, the reactor must be brought to a MODE in which the Required Actions do not apply. The allowed Completion Time of 6 hours to be in MODE 3 is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for any particular RPS Function are found in the SR column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and response time testing.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~In order for a plant to take credit for topical reports as the basis for justifying Frequencies, topical reports must be supported by an NRC staff SER that establishes the acceptability of each topical report for that plant (Ref. 9).~~
~~---~~

~~----------REVIEWER'S NOTE----------~~
~~Notes a and b are applied to the setpoint verification Surveillances for each RPS Instrumentation – Operating (Analog) Function in Table 3.3.1-1 unless one or more of the following exclusions apply:~~

~~1. Manual actuation circuits, automatic actuation logic circuits or instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.~~

~~2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.~~

~~3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.~~
~~---~~ 3

SR 3.3.1.1

Performance of the CHANNEL CHECK ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

~~[The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

SR 3.3.1.2

15 A daily calibration (heat balance) is performed when THERMAL POWER is ≥ ~~20~~%. The daily calibration shall consist of adjusting the "nuclear power calibrate" potentiometers to agree with the calorimetric calculation ~~if the absolute difference is > 1.5%~~. The "ΔT power calibrate" potentiometers are then used to null the "nuclear power - ΔT power" indicators on the RPS Reactor Power Calibration and Indication panel. Performance of the daily calibration ensures that the two inputs to the Q power measurement are indicating accurately with respect to the much more accurate secondary calorimetric calculation. 1 1

~~[The Frequency of 24 hours is based on plant operating experience and takes into account indications and alarms located in the control room to detect deviations in channel outputs.~~ 3

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

15 The Frequency is modified by a Note indicating this Surveillance must be performed within 12 hours after THERMAL POWER is ≥ ~~20~~% RTP. The secondary calorimetric is inaccurate at lower power levels. The 12 hours allows time requirements for plant stabilization, data taking, and instrument calibration. 1

A second Note indicates the ~~daily~~ calibration may be suspended during PHYSICS TESTS. This ensures that calibration is proper preceding and following physics testing at each plateau, recognizing that during testing, changes in power distribution and RCS temperature may render the calorimetric inaccurate. 1 R1

SR 3.3.1.3

It is necessary to calibrate the excore power range channel upper and lower subchannel amplifiers such that the internal ASI used in the TM/LP and ~~APD~~ - High trips reflects the true core power distribution as determined by the incore detectors. A Note to the Frequency indicates the Surveillance is required within 12 hours after THERMAL POWER is ≥ ~~[20]~~% RTP. Uncertainties in the excore and incore measurement process make it impractical to calibrate when THERMAL POWER is < ~~[20]~~% RTP. The Completion Time of 12 hours allows time for plant stabilization, data taking, and instrument calibration. If the excore detectors are not properly calibrated to agree with the incore detectors, power is restricted during subsequent operations because of increased uncertainty associated with using uncalibrated excore detectors. ~~[The 31 day Frequency is adequate, based on operating experience of the excore linear amplifiers and the slow burnup of the detectors. The excore readings are a strong function of the power produced in the peripheral fuel bundles and do not represent an integrated reading across the core. Slow changes in neutron flux during the fuel cycle can also be detected at this Frequency.~~

Local Power Density 1

15 3

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed on each RPS instrument channel, except Loss of Load and Power Rate of Change to ensure the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

In addition to power supply tests, The RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference ~~8~~ 9. These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. They include:

Bistable Tests

The bistable setpoint must be found to trip conservative with respect to the Allowable Values specified in the LCO and left set consistent with the assumptions of the plant specific setpoint analysis (Ref. ~~7~~). As-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference 10. 8 1

acceptance criteria A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified ~~tolerance~~ around the setpoint. This is done with the affected RPS channel trip channel bypassed. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value. 5
Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition.
The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[LTSP]~~. Where a setpoint more conservative than the ~~[LTSP]~~ is used in the plant surveillance procedures ~~[NTSP]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and ~~/or~~ Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[LTSP]~~, then the channel shall be declared inoperable. 5 3 5 1 3 5

The second Note also requires that ~~[LTSP]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~. 3 5

Matrix Logic Tests

Matrix Logic tests are addressed in LCO 3.3.~~3~~2. This test is performed one matrix at a time. It verifies that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. This test will detect any short circuits around the bistable contacts in the coincidence logic, such as may be caused by faulty bistable relay or trip channel bypass contacts. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

Trip Path Tests

Trip Path (Initiation Logic) tests are addressed in LCO 3.3.~~3~~. These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, opening the affected set of RTCBs. The RTCBs must then be closed prior to testing the other three initiation circuits, or a reactor trip may result.

~~[The Frequency of [92] days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 10).~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

SR 3.3.1.5

A CHANNEL CALIBRATION of the excore power range channels ensures that the channels are reading accurately and within ~~tolerance~~. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

~~The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference [10].~~

A Note is added stating that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by

BASES

SURVEILLANCE REQUIREMENTS (continued)

performing the ~~daily~~ calorimetric calibration (SR 3.3.1.2) and the ~~monthly~~ linear subchannel gain check (SR 3.3.1.3). In addition, associated control room indications are continuously monitored by the operators. 1

~~[The Frequency of 92 days is acceptable, based on plant operating experience, and takes into account indications and alarms available to the operator in the control room.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 3

SR 3.3.1.5 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value. 5
Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition.
The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[LTSP]~~. Where a setpoint more conservative than the ~~[LTSP]~~ is used in the plant surveillance procedures ~~[NTSP]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[LTSP]~~, then the channel shall be declared inoperable. 5 5 3 5

acceptance criteria band
acceptance criteria band
NTSP
acceptance criteria bands
acceptance criteria band
NTSP

BASES

SURVEILLANCE REQUIREMENTS (continued)

NTSP

acceptance criteria bands

The second Note also requires that ~~[LTSP]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~.

3 5 R1

the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively

SR 3.3.1.6

A CHANNEL FUNCTIONAL TEST on the Loss of Load and Power Rate of Change channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. ~~The Loss of Load pressure sensor cannot be tested during reactor operation without closing the high pressure TSV, which would result in a turbine trip or reactor trip.~~ The Power Rate of Change - High trip Function is required during startup operation and is bypassed when shut down or > 15% RTP.

The Loss of Load trip Function is not tested during reactor operation because performing this test during operation could cause a pressure perturbation in the emergency trip fluid header common to all four channels resulting in a turbine trip and reactor trip.

1 R1

SR 3.3.1.6 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition.

acceptance criteria band

5

acceptance criteria band

NTSP

acceptance criteria bands

acceptance criteria band

The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[LTSP]~~. Where a setpoint more conservative than the ~~[LTSP]~~ is used in the plant surveillance procedures ~~[NTSP]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[LTSP]~~, then the channel shall be declared inoperable.

5 5 3 5

NTSP

BASES

SURVEILLANCE REQUIREMENTS (continued)

NTSP

acceptance criteria bands

The second Note also requires that ~~[LTSP]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~. 3 5

the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively

SR 3.3.1.7

SR 3.3.1.7 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.1.4, except SR 3.3.1.7 is applicable only to bypass Functions ~~and is performed once within 92 days prior to each startup~~. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Proper operation of bypass permissives is critical during plant startup because the bypasses must be in place to allow startup operation and must be removed at the appropriate points during power ascent to enable certain reactor trips. ~~Consequently, the appropriate time to verify bypass removal function OPERABILITY is just prior to startup. The allowance to conduct this test within 92 days of startup is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 10).~~ Once the operating bypasses are removed, the bypasses must not fail in such a way that the associated trip Function gets inadvertently bypassed. This feature is verified by the trip Function CHANNEL FUNCTIONAL TEST, SR 3.3.1.4. ~~Therefore, further testing of the bypass function after startup is unnecessary.~~

1 R1

1 R1

1 R1

SR 3.3.1.8

SR 3.3.1.8 is the performance of a CHANNEL CALIBRATION.

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference ~~[~~10~~]~~. 3

SR 3.3.1.8 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value 5 (e.g., outside the conservative side of the as-found acceptance criteria band). Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition.

The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[LTSP]~~. Where a setpoint more conservative than the ~~[LTSP]~~ is used in the plant surveillance procedures ~~[NTSP]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[LTSP]~~, then the channel shall be declared inoperable. 5 5 3 5

The second Note also requires that ~~[LTSP]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~ . the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively 5 3

~~[The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4

The Surveillance is modified by a Note to indicate that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the calorimetric calibration (SR 3.3.1.2) and the linear subchannel gain check (SR 3.3.1.3).

SR 3.3.1.9

This SR ensures that the RPS RESPONSE TIMES are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the RTCBs open. ~~[Response times are conducted on an [18] month STAGGERED TEST BASIS. This results in the interval between successive surveillances of a given channel of n x 18 months, where n is the number of channels in the function. The Frequency of [18] months is based upon operating experience, which has shown that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Also, response times cannot be determined at power, since equipment operation is required.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

Testing may be performed in one measurement or in overlapping segments, with verification that all components are tested.

~~----------REVIEWER'S NOTE----------~~
~~Applicable portions of the following TS Bases are applicable to plants adopting CEOG Topical Report CE NPSD-1167-1, "Elimination of Pressure Sensor Response Time Testing Requirements," and the methodology contained in Attachment 1 to TSTF-569.~~
~~----------~~ 4

Response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements," (Ref. ~~1~~1) provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the Topical Report. The response time may be verified for components that replace the components that were previously evaluated in Ref. ~~1~~1 provided that the components have been evaluated in accordance with the NRC approved methodology as discussed in Attachment 1 to TSTF-569, "Methodology to Eliminate Pressure Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing," (Ref. ~~1~~2). Response time verification for other sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and reverified after maintenance that may adversely affect the sensor response time.

10 10 11 1 R1 R1 R1

A Note is added to indicate that the neutron detectors are excluded from RPS RESPONSE TIME testing because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the ~~daily~~ calorimetric calibration (SR 3.3.1.2).

1

REFERENCES

1. Technical Requirements Manual (TRM)
2. ~~1~~. Regulatory Guide 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation."
3. ~~2~~. 10 CFR 50, Appendix A, GDC 21.
4. ~~3~~. 10 CFR ~~100~~ 50.67.
5. ~~4~~. IEEE Standard 279-1971, April 5, 1972.

1

BASES

REFERENCES (continued)

6 ~~5~~. FSAR, Chapter ~~[14]~~ U 15.

7 ~~6~~. 10 CFR 50.49.

8 ~~7~~. ~~"Plant Protection System Selection of Trip Setpoint Values."~~ IC-3.17, "Instrument Setpoint Methodology for Nuclear Power Plants"

9 ~~8~~. FSAR, Section ~~[~~7.2~~]~~. U

~~9. NRC Safety Evaluation Report, [Date].~~

~~10. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989.~~ R1

10 ~~11~~. CEOG Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements."

11 ~~12~~. Attachment 1 to TSTF-569, "Methodology to Eliminate Pressure Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing."

3 3 1

BASES

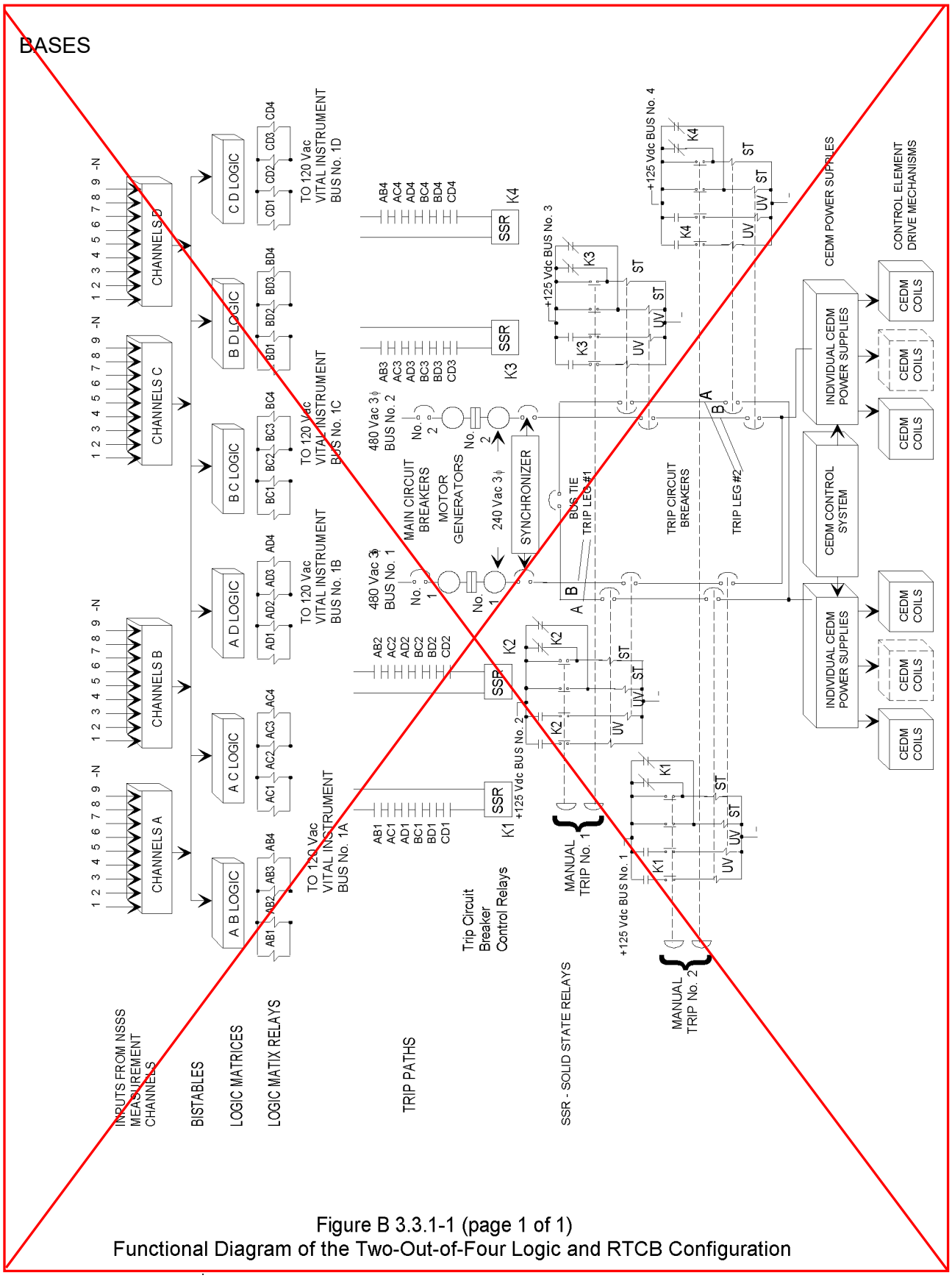

Figure B 3.3.1-1 (page 1 of 1)
Functional Diagram of the Two-Out-of-Four Logic and RTCB Configuration

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.1 Reactor Protective System (RPS) Instrumentation ~~– Operating (Analog)~~ 2

BASES

BACKGROUND

The Reactor Protective System (RPS) initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary (RCPB) during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulation as "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for channel uncertainties related to the setting at which the automatic protective action would actually occur.

~~---------- REVIEWER'S NOTE ----------~~

~~The term "Limiting Trip Setpoint" [LTSP] is generic terminology for the calculated trip setting (setpoint) value calculated by means of the plant specific setpoint methodology documented in a document controlled under 10 CFR 50.59. The term [LTSP] indicates that no additional margin has been added between the Analytical Limit and the calculated trip setting.~~

~~"Nominal Trip Setpoint [NTSP]" is the suggested terminology for the actual setpoint implemented in the plant surveillance procedures where margin has been added to the calculated [LTSP]. The as-found and as-left tolerances will apply to the [NTSP] implemented in the Surveillance procedures to confirm channel performance.~~ 4

BASES

BACKGROUND (continued)

~~Licensees are to insert the name of the document(s) controlled under 10 CFR 50.59 that contain the methodology for calculating the as-left and as-found tolerances in Note b of Table 3.3.1-1 for the phrase "[insert the name of a document controlled under 10 CFR 50.59 such as the Technical Requirements Manual or any document incorporated into the facility FSAR]" throughout these Bases.~~

~~Where the [LTSP] is not included in Table 3.3.1-1, the plant specific location for the [LTSP] or [NTSP] must be cited in Note b of Table 3.3.1-1. The brackets indicate plant specific terms may apply, as reviewed and approved by the NRC.~~

4

Nominal NTSP Reference 1

The ~~[Limiting~~ Trip Setpoint (~~LTSP~~)~~]~~ specified in ~~Table 3.3.1-1~~ is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the ~~[LTSP]~~ accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the ~~[LTSP]~~ ensures that SLs are not exceeded. Therefore, the ~~[LTSP]~~ meets the definition of an LSSS (Ref. ~~1~~).

1 3 1

NTSP 2

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety function(s)." Relying solely on the ~~[LTSP]~~ to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a Surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the ~~[LTSP]~~ due to some drift of the setting may still be OPERABLE because drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the ~~[LTSP]~~ and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE because it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left ~~tolerance~~ around the ~~[LTSP]~~ to account for further drift during the next surveillance interval. Note that,

NTSP

3

acceptance criteria band NTSP 5

BASES

BACKGROUND (continued)

acceptance criteria band

although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left ~~tolerance~~ in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria). 5

However, there is also some point beyond which the channel may not be able to perform its function due to, for example, greater than expected drift. This value needs to be specified in the Technical Specifications in order to define OPERABILITY of the channels and is designated as the Allowable Value.

acceptance criteria band (e.g., conservative side of the as-found acceptance criteria band)

If the actual setting (as-found setpoint) of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found ~~tolerance band~~, the channel is OPERABLE, but degraded. The degraded condition will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the ~~[Nominal Trip Setpoint (~~NTSP~~)]~~ (within the allowed ~~tolerance~~), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation. 5 3

acceptance criteria band

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling,
- Fuel centerline melting shall not occur, and
- The Reactor Coolant System (RCS) pressure SL of ~~[~~2750~~]~~ psia shall not be exceeded. 3

Maintaining the parameters within the above values ensures that the offsite dose will be within the ~~10 CFR 50 (Ref. 2) and~~ 10 CFR ~~100~~ (Ref. ~~3~~) criteria during AOOs. 1

4

50.67

BASES

BACKGROUND (continued)

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~ (Ref. ~~3~~) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event. 1

50.67

4

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels,
- Bistable trip units,
- RPS Logic, and
- Reactor trip circuit breakers (RTCBs).

This LCO addresses measurement channels and bistable trip units. It also addresses the automatic bypass removal feature for those trips with operating bypasses. The RPS Logic and RTCBs are addressed in LCO 3.3.~~3~~, "Reactor Protective System (RPS) Logic and Trip Initiation." 1

2

The role of each of these modules in the RPS, including those associated with the logic and RTCBs, is discussed below.

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

The excore nuclear instrumentation and the analog core protection calculators (CPCs) are considered components in the measurement channels. The wide range nuclear instruments (NIs) provide a Power Rate of Change - High Trip. Three RPS trips use a power level designated as Q power as an input. Q power is the higher of NI power and primary calorimetric power (ΔT power) based on RCS hot leg and cold leg temperatures. Trips using Q power as an input include the Variable ~~High~~ Power Trip (VHPT) - High, Thermal Margin/Low Pressure (TM/LP), and the ~~Axial~~ Power ~~Distribution (APD)~~ - High trips. 1

Level

Local

Density

BASES

BACKGROUND (continued)

Local Power Density

The analog CPCs provide the complex signal processing necessary to calculate the TM/LP trip setpoint, ~~APD~~ trip setpoint, VHPT trip setpoint, and Q power calculation. 1

linear

Local Power Density

The excore NIs (wide range and ~~power~~ range) and the analog CPCs (TM/LP and ~~APD~~ calculators) are mounted in the RPS cabinet, with one channel of each in each of the four RPS bays. 1

Four identical measurement channels, designated channels A through D, with electrical and physical separation are provided for each parameter used in the direct generation of trip signals. Measurement channels provide input to one or more RPS bistables within the same RPS channel. In addition, some measurement channels may also be used as inputs to Engineered Safety Features Actuation System (ESFAS) bistables, and most provide indication in the control room. Measurement channels used as an input to the RPS are never used for control functions.

When a channel monitoring a parameter exceeds a predetermined setpoint, indicating an unsafe condition, the bistable monitoring the parameter in that channel will trip. Tripping two or more channels of

the

bistables monitoring the same parameter de-energizes Matrix Logic, which in turn de-energizes the Initiation Logic. This causes all eight RTCBs to open, interrupting power to the control element assemblies (CEAs), allowing them to fall into the core. 7

Three of the four measurement and bistable channels are necessary to meet the redundancy and testability of GDC 21 in 10 CFR 50, Appendix A (Ref. ~~2~~). The fourth channel provides additional flexibility by allowing one channel to be removed from service (trip channel bypass) for maintenance or testing while still maintaining a minimum two-out-of-three logic. Thus, even with a channel inoperable, no single additional failure in the RPS can either cause an inadvertent trip or prevent a required trip from occurring. 1

3

Since no single failure will either cause or prevent a protective system actuation, and no protective channel feeds a control channel, this arrangement meets the requirements of IEEE Standard 279-1971 (Ref. 4). 1

5

NTSP

Many of the RPS trips are generated by comparing a single measurement to a fixed bistable ~~[LTSP]~~. Certain Functions, however, make use of more than one measurement to provide a trip. The following trips use multiple measurement channel inputs: 3

BASES

BACKGROUND (continued)

- Steam Generator Level - Low

  This trip uses the lower of the two steam generator levels as an input to a common bistable.

- Steam Generator Pressure - Low

  This trip uses the lower of the two steam generator pressures as an input to a common bistable.

- Variable ~~High~~ Power Trip (VHPT) - High (Level) 1

  The VHPT uses Q power as its only input. Q power is the higher of NI power and ΔT power. It has a trip setpoint that tracks power levels downward so that it is always within a fixed increment above current power, subject to a minimum value.

  On power increases, the trip setpoint remains fixed unless manually reset, at which point it increases to the new setpoint, a fixed increment above Q power at the time of reset, subject to a maximum value. Thus, during power escalation, the trip setpoint must be repeatedly reset to avoid a reactor trip.

- Thermal Margin/Low Pressure (TM/LP) and Steam Generator Pressure Difference

  Q power is only one of several inputs to the TM/LP trip. Other inputs include internal ASI and cold leg temperature based on the higher of two cold leg resistance temperature detectors. The TM/LP trip setpoint is a complex function of these inputs and represents a minimum acceptable RCS pressure to be compared to actual RCS pressure in the TM/LP trip unit.

  Steam generator pressure is also an indirect input to the TM/LP trip via the Steam Generator Pressure Difference. This Function provides a reactor trip when the secondary pressure in either steam generator exceeds that of the other generator by greater than a fixed amount. The trip is implemented by biasing the TM/LP trip setpoint upward so as to ensure TM/LP trip if an asymmetric steam generator transient is detected.

- ~~Axial Power Distribution (APD)~~ – High

  Q Power and ASI are inputs to the ~~APD~~ trip. The ~~APD~~ trip setpoint is a function of Q power, being more restrictive at higher power levels. It provides a reactor trip if actual ASI exceeds the ~~APD~~ trip setpoint. (Local Power Density) (Local Power Density) 1

BASES

BACKGROUND (continued)

Bistable Trip Units

Bistable trip units, mounted in the RPS cabinet, receive an analog input from the measurement channels, compare the analog input to trip setpoints, and provide contact output to the Matrix Logic. They also provide local trip indication and remote annunciation.

There are four channels of bistable trip units, designated A through D, for each RPS Function, one for each measurement channel. Bistable output relays de-energize when a trip occurs.

The contacts from these bistable relays are arranged into six coincidence matrices, comprising the Matrix Logic. If bistables monitoring the same parameter in at least two channels trip, the Matrix Logic will generate a reactor trip (two-out-of-four logic).

Some of the RPS measurement channels provide contact outputs to the RPS, so the comparison of an analog input to a trip setpoint is not necessary. In these cases, the bistable trip unit is replaced with an auxiliary trip unit. The auxiliary trip units provide contact multiplication so the single input contact opening can provide multiple contact outputs to the coincidence logic as well as trip indication and annunciation.

Local Power Density

turbine generator control system hydraulic

Local Power Density

Trips employing auxiliary trip units include the Loss of Load trip and the ~~APD~~ - High trip. The Loss of Load trip is a contact input from the ~~Electro Hydraulic Control System~~ control oil pressure ~~on each of the~~ four high pressure ~~stop valves~~.

switches mounted on the emergency trip fluid line common header

sensed by

1

R1

The ~~APD~~ trip, described above, is a complex function in which the actual trip comparison is performed within the CPC. Therefore the ~~APD~~ - High trip unit employs a contact input from the CPC.

1

Local Power Density

All RPS trips, with the exception of the Loss of Load trip, generate a pretrip alarm as the trip setpoint is approached.

6

The trip setpoints used in the bistable trip units are based on the analytical limits stated in Reference ~~5~~. The calculation of the ~~[LTSP]~~ specified in Table 3.3.1-1 is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration ~~tolerances~~, instrumentation uncertainties, instrument drift, and severe environment errors - for those RPS channels that must function in harsh environments, as defined by 10 CFR 50.49 (Ref. ~~6~~) - Allowable Values specified in Table 3.3.1-1, in the accompanying LCO,

NTSP

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acceptance criteria bands

1

5

7

BASES

BACKGROUND (continued)

are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in ~~the "Plant Protection System Selection of Trip Setpoint Values"~~ (Ref. ~~7~~). The trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value, to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the interval between surveillances.

IC-3.17, "Instrument Setpoint Methodology for Nuclear Power Plants" 8 1

The ~~[~~LTSP~~]~~ is the value at which the bistable is set and is the expected value to be achieved during calibration. The ~~[~~LTSP~~]~~ value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties.

NTSP 3

~~[~~LTSPs~~]~~, in conjunction with the use of as-found and as-left ~~tolerances~~, consistent with the requirements of the Allowable Value will ensure that SLs of Chapter 2.0 are not violated during AOOs and the consequences of DBAs will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

acceptance criteria bands 3 5

Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the least conservative value of the as-found setpoint that a channel can have during a periodic CHANNEL CALIBRATION or CHANNEL FUNCTIONAL TEST.

RPS Logic

The RPS Logic, addressed in LCO 3.3.~~3~~, consists of both Matrix and Initiation Logic and employs a scheme that provides a reactor trip when bistables in any two out of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic. ~~This logic and the RTCB configuration are shown in Figure B 3.3.1-1.~~

2 1 6

Bistable relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

BASES

BACKGROUND (continued)

The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized RTCB control relays (K1, K2, K3, and K4). The trip paths thus each have six contacts in series, one from each matrix, and perform a logical OR function, opening the RTCBs if any one or more of the six logic matrices indicate a coincidence condition.

Each trip path is responsible for opening one set of two of the eight RTCBs. The RTCB control relays (K-relays), when de-energized, interrupt power to the breaker undervoltage trip attachments and simultaneously apply power to the shunt trip attachments on each of the two breakers. Actuation of either the undervoltage or shunt trip attachment is sufficient to open the RTCB and interrupt power from the motor generator (MG) sets to the control element drive mechanisms (CEDMs).

When a coincidence occurs in two RPS channels, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four RTCB control relays, which simultaneously de-energize the undervoltage and energize the shunt trip attachments in all eight RTCBs, tripping them open.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting matrix wiring between bistable and auxiliary trip units, up to but not including the matrix relays. Contacts in the bistable and auxiliary trip units are excluded from the Matrix Logic definition, since they are addressed as part of the measurement channel.

The Initiation Logic consists of the trip path power source, matrix relays and their associated contacts, all interconnecting wiring, and solid state (auxiliary) relays through the K-relay contacts in the RTCB control circuitry.

It is possible to change the two-out-of-four RPS Logic to a two-out-of-three logic for a given input parameter in one channel at a time by trip channel bypassing select portions of the Matrix Logic. Trip channel bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional channels indicate a trip condition. Trip channel bypassing can be simultaneously performed on

BASES

BACKGROUND (continued)

A mechanical interlock (key capture) in conjunction with administrative control

any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. ~~An interlock~~ prevents simultaneous trip channel bypassing of the same parameter in more than one channel. Trip channel bypassing is normally employed during maintenance or testing. 1

For those plants that have demonstrated sufficient channel to channel independence, two-out-of-three logic is the minimum that is required to provide adequate plant protection, since a failure of one channel still ensures a reactor trip would be generated by the two remaining OPERABLE channels. Two-out-of-three logic also prevents inadvertent trips caused by any single channel failure in a trip condition.

Zero Power Mode trips (Reactor Coolant Flow – Low and Thermal Margin/Low Pressure), Local Power Density - High, Loss of Load -High,

Local Power Density

In addition to the trip channel bypasses, there are also operating bypasses on select RPS trips. Some of these bypasses are enabled manually, others automatically, in all four RPS channels when plant conditions do not warrant the specific trip protection. All operating bypasses are automatically removed when enabling bypass conditions are no longer satisfied. Trips with operating bypasses include Power Rate of Change - High, ~~Reactor Coolant Flow - Low,~~ Steam Generator Pressure - Low, ~~APD - High, TM/LP,~~ and Steam Generator Pressure Difference. ~~[~~The Loss of Load trip, Power Rate of Change - High, and ~~APD~~ - High operating bypasses are automatically enabled and disabled.~~]~~ 1 3 R1

The Variable Power Level Trip – High bypass is manual enable but automatically

Reactor Trip Circuit Breakers (RTCBs)

2

The reactor trip switchgear, addressed in LCO 3.3.~~3 and shown in Figure B 3.3.1-1~~, consists of eight RTCBs, which are operated in four sets of two breakers (four channels). Power input to the reactor trip switchgear comes from two full capacity MG sets operated in parallel such that the loss of either MG set does not de-energize the CEDMs. There are two separate CEDM power supply buses, each bus powering half of the CEDMs. Power is supplied from the MG sets to each bus via two redundant paths (trip legs). Trip legs 1A and 1B supply power to CEDM bus 1. Trip legs 2A and 2B supply power to CEDM bus 2. This ensures that a fault or the opening of a breaker in one trip leg (i.e., for testing purposes) will not interrupt power to the CEDM buses. 6

Each of the four trip legs consists of two RTCBs in series. The two RTCBs within a trip leg are actuated by separate initiation circuits.

BASES

BACKGROUND (continued)

The eight RTCBs are operated as four sets of two breakers (four channels). For example, if a breaker receives an open signal in trip leg A (for CEDM bus 1), an identical breaker in trip leg B (for CEDM bus 2) will also receive an open signal. This arrangement ensures that power is interrupted to both CEDM buses, thus preventing trip of only half of the CEAs (a half trip). Any one inoperable breaker in a channel will make the entire channel inoperable.

Each set of RTCBs is operated by either a Manual Trip push button or an RPS actuated K-relay. There are four Manual Trip push buttons~~,~~ arranged in two sets of two~~, as shown in Figure B 3.3.1-1~~. Depressing both push buttons in either set will result in a reactor trip. 6

When a Manual Trip is initiated using the control room push buttons, the RPS trip paths and K-relays are bypassed, and the RTCB undervoltage and shunt trip attachments are actuated independent of the RPS.

Manual Trip circuitry includes the push button and interconnecting wiring to both RTCBs necessary to actuate both the undervoltage and shunt trip attachments but excludes the K-relay contacts and their interconnecting wiring to the RTCBs, which are considered part of the Initiation Logic.

Functional testing of the entire RPS, from bistable input through the opening of individual sets of RTCBs, can be performed either at power or shutdown and is ~~normally~~ performed ~~on a quarterly basis~~. per the Surveillance Frequency Control Program 9 1 3
FSAR, Section ~~[~~7.2~~]~~ (Ref. ~~3~~), explains RPS testing in more detail. U

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Reference ~~5~~ takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis are part of the NRC approved licensing basis for the plant. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. Other Functions, such as the Loss of Load trip, are purely equipment protective, and their use minimizes the potential for equipment damage. 6 1

Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The specific safety analyses applicable to each protective Function are identified below:

1. Variable ~~High~~ Power Level Trip (VHPT) - High 1

The VHPT provides reactor core protection against positive reactivity excursions that are too rapid for a Pressurizer Pressure - High or TM/LP trip to protect against. The following events require VHPT protection:

- Uncontrolled CEA withdrawal event,
- Excess load, R1
- Excess feedwater heat removal event,
- CEA ejection event, and
-
- Main steam line break (MSLB) ~~(outside containment)~~. R1

The first three events are AOOs, and fuel integrity is maintained. The fourth and fifth are accidents, and limited fuel damage may occur.

2. Power Rate of Change - High

The Power Rate of Change - High trip is used to trip the reactor when excore ~~[~~logarithmic~~]~~ power indicates an excessive rate of change. The Power Rate of Change - High Function ~~minimizes transients for events such as a continuous CEA withdrawal or a boron dilution event from low power levels~~. The trip may be bypassed when THERMAL POWER is < 1E-4% RTP, when poor counting statistics may lead to erroneous indication. ~~It is also~~ bypassed at > 15% RTP, where moderator temperature coefficient and fuel temperature coefficient make high rate of change of power unlikely. With the RTCBs open, the Power Rate of Change - High trip is not required to be OPERABLE; however, the indication ~~and alarm~~ Function~~s~~ of at least two channels ~~are~~ is required by LCO 3.3.1~~3~~, "~~[~~Logarithmic~~]~~ ~~Power~~ Monitoring ~~Channels~~," to be OPERABLE. LCO 3.3.1~~3~~ ensures the ~~[~~logarithmic~~]~~ channels are available to detect ~~and alert the operator to~~ a boron dilution event.

is not credited in the accident analysis as providing primary protection for any limiting case AOO or accident.

In addition, the trip is not required to be OPERABLE and is

Neutron Flux

3 1 1 3 1 1 3 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

3. Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection during the following events:

- Loss of RCS flow,
- Loss of ~~nonemergency~~ AC power,
- Reactor coolant pump (RCP) seized shaft, and
- RCP sheared shaft~~, and~~
- ~~Certain MSLB events~~.

R1

The loss of RCS flow and ~~of nonemergency~~ AC power events are AOOs where fuel integrity is maintained. The RCP seized shaft, and sheared shaft~~, and MSLBs~~ are accidents where fuel damage may result.

4. Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with pressurizer safety valves and main steam safety valves (MSSVs), provides protection against overpressure conditions in the RCS during the following events:

1

- Loss of condenser vacuum with a concurrent loss of offsite power,
- Loss of condenser vacuum with a concurrent loss of one 6.9 kV bus,
- ~~Isolation of turbine at 102% power~~ Turbine trip,
- Feedwater ~~System pipe breaks between the steam generator and check valve~~ line break (small),
- CEA withdrawal, and
- Loss of feedwater flow.

R1

BASES

APPLICABLE SAFETY ANALYSES (continued)

5. Containment Pressure - High

The Containment Pressure - High trip prevents exceeding the containment design pressure during certain loss of coolant accidents (LOCAs) or feedwater line break accidents. It ensures a reactor trip prior to, or concurrent with, a LOCA, thus assisting the ESFAS in the event of a LOCA or MSLB. Since these are accidents, SLs may be violated. However, the consequences of the accident will be acceptable.

6. Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection against an excessive rate of heat extraction from the steam generators, which would result in a rapid uncontrolled cooldown of the RCS. This trip is needed to shut down the reactor and assist the ESFAS in the event of an MSLB. Since these are accidents, SLs may be violated. However, the consequences of the accident will be acceptable.

7.a, 7.b. Steam Generator A and B Level - Low

The Steam Generator A Level - Low and Steam Generator B Level - Low trips are required for the following events:

- Steam System piping failures,
- Feedwater System pipe breaks,
- ~~Inadvertent opening of a steam generator atmospheric dump valve (ADV),~~ Excess Load and
- Loss of normal feedwater~~, and~~
- ~~Asymmetric loss of feedwater~~.

R1 1

The Steam Generator Level - Low trip ensures that low DNBR, high local power density, and the RCS pressure SLs are maintained during normal operation and AOOs, and, in conjunction with the ESFAS, the consequences of the Feedwater System pipe break accident will be acceptable.

BASES

APPLICABLE SAFETY ANALYSES (continued)

8. ~~Axial Power Distribution (APD)~~ - High

Local Power Density

1

The ~~APD~~ - High trip ensures that excessive axial peaking, such as that due to axial xenon oscillations, will not cause fuel damage. It ensures that neither a DNBR less than the SL nor a peak linear heat rate that corresponds to the temperature for fuel centerline melting will occur. This trip is the primary protection against fuel centerline melting.

9. Thermal Margin

a. Thermal Margin/Low Pressure (TM/LP)

The TM/LP trip prevents exceeding the DNBR SL during AOOs and aids the ESFAS during certain accidents. The following events require TM/LP protection:

- Excess load ~~(inadvertent opening of a steam generator ADV)~~, R1
- RCS depressurization (inadvertent safety or power operated relief valves (PORVs) opening), R1
- Steam generator tube rupture, and
- LOCA accident.

The first two events are AOOs, and fuel integrity is maintained. The third and fourth are accidents, and limited fuel damage may occur although only the LOCA is expected to result in fuel damage. The trip is initiated whenever the RCS pressure signal drops below a minimum value (P_{min}) or a computed value (P_{var}) as described below, whichever is higher. The computed value is a Function Q power, ASI, as determined from the axially split excore detectors~~,~~ and reactor inlet (cold leg) temperature~~, and the number of RCPs operating~~.

and

1

The minimum value of reactor coolant flow rate, the maximum ~~T~~$_Q$, and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip Function. In addition, CEA group sequencing in accordance with LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits," is assumed. Finally, the maximum insertion of CEA banks that can occur during any AOO prior to a VHPT is assumed.

azimuthal tilt

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. Steam Generator Pressure Difference

The Steam Generator Pressure Difference provides protection for those AOOs associated with secondary system malfunctions that result in asymmetric primary coolant temperatures. The most limiting event is closure of a single main steam isolation valve. Steam Generator Pressure Difference is provided by comparing the secondary pressure in both steam generators in the TM/LP calculator. If the pressure in either exceeds that in the other by the trip setpoint, a TM/LP trip will result.

10. Loss of Load

Hydraulic Fluid Pressure - Low

R1

1

The Loss of Load (turbine ~~stop valve (TSV)~~ control oil pressure) trip is anticipatory for the loss of heat removal capabilities of the secondary system following a turbine trip. The Loss of Load trip prevents lifting the pressurizer safety valves, PORVs, and MSSVs in the event of a turbine generator trip. Thus, the trip minimizes the pressure and temperature transients on the reactor by initiating a trip well before reaching the Pressurizer Pressure - High trip and pressurizer safety valve setpoints. The four RPS Loss of Load reactor trip channels receive their input from ~~sensors mounted on the high pressure TSV actuators. Since there are four high pressure TSVs, one actuator per valve and one sensor per actuator, each sensor sends its signal to a different RPS channel. When the turbine trips, control oil is dumped from the high pressure TSVs. When the control oil pressure drops to the appropriate setpoint, a reactor trip signal is generated.~~

four redundant pressure switches mounted on the emergency trip fluid line common header, to serve as the loss of load, turbine trip input to the Reactor Protective System logic matrices. Actuation of any two of the pressure switches on low hydraulic oil pressure causes a reactor trip.

1

Interlocks/Bypasses

The bypasses and their Allowable Values are addressed in footnotes to Table 3.3.1-1. They are not otherwise addressed as specific Table entries.

The automatic bypass removal features must function as a backup to manual actions for all safety related trips to ensure the trip Functions are not operationally bypassed when the safety analysis assumes the Functions are not bypassed. The RPS operating bypasses are:

Zero power mode bypass (ZPMB) removal on the TM/LP, Steam Generator Pressure Difference, and reactor coolant low flow trips when THERMAL POWER is < ~~1E-4~~ 0.5% RTP. This bypass is manually enabled below the specified setpoint to permit low power testing. The wide range NI Level 1 bistable in the wide range drawer permits manual bypassing below the setpoint and removes the bypass above the setpoint.

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

Power rate of change bypass removal. The Power Rate of Change - High trip is automatically bypassed at < 1E-4% RTP, as sensed by the wide range NI Level 2 bistable, and at > ~~12~~15% RTP by the power range NI Level 1 bistable, mounted in their respective NI drawers. Automatic bypass removal is also effected by these bistables when conditions are no longer satisfied. 1

Local Power Density

Loss of Load and ~~APD~~ - High bypass removal. The Loss of Load and ~~APD~~ - High trips are automatically bypassed when at < 15% RTP as sensed by the power range NI Level 1 bistable. The bypass is automatically removed by this bistable above the setpoint. This same bistable is used to bypass the Power Rate of Change - High trip. 1

Steam Generator Pressure - Low bypass removal. The Steam Generator Pressure - Low trip is manually enabled below the pretrip setpoint. The permissive is removed, and the bypass automatically removed, when the Steam Generator Pressure - Low pretrip clears.

The RPS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The specific criteria for determining channel OPERABILITY differ slightly between Functions. These criteria are discussed on a Function by Function basis below.

Actions allow maintenance (trip channel) bypass of individual channels, but ~~the bypass activates interlocks that prevent~~ operation with a second channel in the same Function bypassed~~.~~. ~~Plants are restricted to 48 hours in a trip channel bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic). At plants where adequate channel to channel independence has been demonstrated, specific exceptions may be approved by the NRC staff to permit one of the two-out-of-four channels to be bypassed for an extended period of time.~~ R1 1

is prevented by mechanical interlock (key capture)

the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively

NTSPs

Allowable Values for RPS Instrumentation Functions are specified in Table 3.3.1-1. ~~[~~LTSPs~~]~~ and the methodologies for calculation of the as-left and as-found ~~tolerances~~ are described in ~~[insert the name of a document controlled under 10 CFR 50.59 such as the Technical Requirements Manual or any document incorporated into the facility FSAR]~~. The ~~[~~LTSPs~~]~~ are selected to ensure that the actual setpoints remain conservative with respect to the as-found ~~tolerance~~ band between successive CHANNEL CALIBRATIONS. After each calibration the trip setpoint shall be left within the as-left band around the ~~[~~LTSP~~]~~. 5 3 5

NTSPs

acceptance criteria

NTSP

BASES

LCO (continued)

The following Bases for each trip Function identify the above RPS trip Function criteria items that are applicable to establish the trip Function OPERABILITY.

Level

1. Variable ~~High~~ Power Trip (VHPT) - High 1

This LCO requires all four channels of the VHPT to be OPERABLE in MODES 1 and 2.

The Allowable Value is high enough to provide an operating envelope that prevents unnecessary Linear Power Level - High reactor VHPT - High trips during normal plant operations. The Allowable Value is low enough for the system to maintain a margin to unacceptable fuel cladding damage should a CEA ejection accident occur.

The VHPT setpoint is operator adjustable and can be set at a fixed increment above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The ~~[LTSP]~~ has a maximum and a minimum setpoint. 3

NTSP

NTSP

Adding to this maximum value the possible variation in ~~[LTSP]~~ due to calibration and instrument errors, the maximum actual steady state THERMAL POWER level at which a trip would be actuated is ~~112~~% RTP, which is the value used in the safety analyses. 3 R1

107

conservative with respect to

of 112% RTP

1

Calibration and instrument errors are considered within the setpoint calculations. The stated minimum and maximum Allowable Values and step values are the assumed safety analysis values.

~~To account for these errors, the safety analysis minimum value is 40% RTP. The 10% step is a maximum value assumed in the safety analysis. There is no uncertainty applied to the step.~~ R1

2. Power Rate of Change - High

MODE 1 with THERMAL POWER ≤ 15% RTP and MODE 2

This LCO requires four channels of Power Rate of Change - High to be OPERABLE in ~~MODES 1 and 2, as well as in MODES 3, 4, and 5 when the RTCBs are closed and the CEA Drive System is capable of CEA withdrawal~~. 1

The high power rate of change trip serves as a backup to the administratively enforced startup rate limit. The Function is not credited in the accident analyses; therefore, the Allowable Value for the trip or bypass Functions is not derived from analytical limits.

BASES

LCO (continued)

3. Reactor Coolant Flow - Low

This LCO requires four channels of Reactor Coolant Flow - Low to be OPERABLE in MODES 1 and 2.

0.5 The trip may be manually bypassed when THERMAL POWER falls below ~~1E-4~~% RTP. This bypass is part of the ZPMB circuitry, which also bypasses the TM/LP trip and provides a ΔT power block signal to the Q power select logic. This ZPMB allows low power physics testing at reduced RCS temperatures and pressures. It also allows heatup and cooldown with shutdown CEAs withdrawn.

This trip is set high enough to maintain fuel integrity during a loss of flow condition. The setting is low enough to allow for normal operating fluctuations from offsite power. ~~To account for analysis uncertainty, the value in the safety analysis is 93% RTP.~~

4. Pressurizer Pressure - High

This LCO requires four channels of Pressurizer Pressure - High to be OPERABLE in MODES 1 and 2.

The Allowable Value is set high enough to allow for pressure increases in the RCS during normal operation (i.e., plant transients) not indicative of an abnormal condition. The setting is below the lift setpoint of the pressurizer safety valves and low enough to initiate a reactor trip when an abnormal condition is indicated. ~~The difference between the Allowable Value and the analysis setpoint of 2470 psia includes allowance for harsh environment.~~

1

The Pressurizer Pressure - High trip concurrent with PORV operation avoids unnecessary operation of the pressurizer safety valves.

5. Containment Pressure - High

This LCO requires four channels of Containment Pressure - High to be OPERABLE in MODES 1 and 2.

The Allowable Value is high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) that are not indicative of an abnormal condition. The setting is low enough to initiate a reactor trip to prevent containment pressure from exceeding design pressure following a DBA.

BASES

LCO (continued)

6. Steam Generator Pressure - Low

This LCO requires four channels of Steam Generator Pressure - Low per steam generator to be OPERABLE in MODES 1 and 2.

The Allowable Value is sufficiently below the full load operating value for steam pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of excessive steam demand. Since excessive steam demand causes the RCS to cool down, resulting in positive reactivity addition to the core, a reactor trip is required to offset that effect.

~~The difference between the Allowable Value and the safety analysis value of 600 psia includes harsh environment uncertainties.~~ 1

The Function may be manually bypassed as steam generator pressure is reduced during controlled plant shutdowns. This bypass is permitted at a preset steam generator pressure. The bypass, in conjunction with the ZPMB, allows testing at low temperatures and pressures, and heatup and cooldown with the shutdown CEAs withdrawn. From a bypass condition the trip will be reinstated automatically as steam generator pressure increases above the preset pressure.

7.a, 7.b. Steam Generator Level – Low

This LCO requires four channels of Steam Generator Level - Low per steam generator to be OPERABLE in MODES 1 and 2.

The Allowable Value is sufficiently below the normal operating level for the steam generators so as not to cause a reactor trip during normal plant operations. The trip setpoint is high enough to ensure a reactor trip signal is generated before water level drops below the top of the feed ring. ~~The difference between the Allowable Value and the measurement value includes 10 inches of measurement uncertainty. The specified setpoint ensures there will be sufficient water inventory to provide a 10 minute margin before auxiliary feedwater is required for the removal of decay heat.~~ 1

BASES

LCO (continued)

8. ~~Axial Power Distribution (APD)~~ - High

Local Power Density

This LCO requires four channels of ~~APD~~ - High to be OPERABLE in MODE 1 ≥ 15% RTP.

The Allowable Value curve was derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore ASI relationship. 1

Local Power Density

The ~~APD~~ trip is automatically bypassed at < 15% RTP, where it is not required for reactor protection.

9. Thermal Margin

a. Thermal Margin/Low Pressure (TM/LP)

This LCO requires four channels of TM/LP to be OPERABLE in MODES 1 and 2.

The Allowable Value includes allowances for equipment response time, measurement uncertainties, processing error, and a further allowance to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the SL.

0.5

This trip may be manually bypassed when THERMAL POWER falls below ~~1E-4~~% RTP. This bypass is part of the ZPMB circuitry, which also bypasses the Reactor Coolant Flow - Low trip and provides a ΔT power block signal to the Q power select logic. This ZPMB allows low power physics testing at reduced RCS temperatures and pressures. It also allows heatup and cooldown with shutdown CEAs withdrawn. 1

b. Steam Generator Pressure Difference

This LCO requires four channels of Steam Generator Pressure Difference to be OPERABLE in MODES 1 and 2.

The Allowable Value is high enough to avoid trips caused by normal operation and minor transients, but ensures DNBR protection in the event of Design Basis Events. ~~The difference between the Allowable Value and the 175 psia analysis setpoint allows for 40 psia of measurement uncertainty.~~ 1

BASES

LCO (continued)

0.5 The trip may be bypassed when THERMAL POWER falls below ~~1E-4~~% RTP. The Steam Generator Pressure Difference is subject to the ZPMB, since it is an input to the TM/LP trip and is not required for protection at low power levels. 1

10. Loss of Load

The LCO requires four Loss of Load trip channels to be OPERABLE in MODE 1 ≥ 15% RTP.

The Loss of Load trip may be bypassed when THERMAL POWER falls below 15%, since it is no longer needed to prevent lifting of the pressurizer safety valves, steam generator safety valves, or PORVs in the event of a Loss of Load. The Nuclear Steam Supply System and the Steam Dump System are capable of accommodating the Loss of Load without requiring the use of the above equipment.

Interlocks/Bypasses

The LCO on bypass permissive removal channels requires that the automatic bypass removal feature of all four operating bypass channels be OPERABLE for each RPS Function with an operating bypass in the MODES addressed in the specific LCO for each Function. All four bypass removal channels must be OPERABLE to ensure that none of the four RPS channels are inadvertently bypassed.

The LCO applies to the bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue.

The interlock Allowable Values are based on analysis requirements for the bypassed functions. These are discussed above as part of the LCO discussion for the affected Functions.

APPLICABILITY This LCO is applicable in accordance with Table 3.3.1-1. Most RPS trips are required to be OPERABLE in MODES 1 and 2 because the reactor is critical in these MODES. The trips are designed to take the reactor subcritical, maintaining the SLs during AOOs and assisting the ESFAS in providing acceptable consequences during accidents. Exceptions are addressed in footnotes to the table. Exceptions to this APPLICABILITY are:

Local Power Density

- The ~~APD~~ - High Trip and Loss of Load are only applicable in MODE 1 ≥ 15% RTP because they may be automatically bypassed at < 15% RTP, where they are no longer needed. 1

BASES

APPLICABILITY (continued)

- ~~The Power Rate of Change - High trip, RPS Logic, RTCBs, and Manual Trip are also required in MODES 3, 4, and 5, with the RTCBs closed, to provide protection for boron dilution and CEA withdrawal events. The Power Rate of Change - High trip in these lower MODES is addressed in LCO 3.3.2, "Reactor Protective System (RPS) Instrumentation - Shutdown." The RPS Logic in MODES 1, 2, 3, 4, and 5 is addressed in LCO 3.3.3.~~

1

The RPS

~~Most~~ trips are not required to be OPERABLE in MODES 3, 4, and 5. In MODES 3, 4, and 5, the emphasis is placed on return to power events. The reactor is protected in these MODES by ensuring adequate SDM.

ACTIONS

acceptance criteria band

The most common causes of channel inoperability are outright failure or drift of the bistable or process module sufficient to exceed the ~~tolerance~~ allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the trip setpoint is nonconservative with respect to the Allowable Value in Table 3.3.1-1, the channel is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.

5

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the channel is not functioning as required, or the transmitter, instrument loop, signal processing electronics, or RPS bistable trip unit is found inoperable, then all affected Functions provided by that channel must be declared inoperable, and the plant must enter the Condition for the particular protection Function affected.

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Times of each inoperable Function will be tracked separately for each Function, starting from the time the Condition was entered.

BASES

ACTIONS (continued)

A.1~~, A.2.1,~~ and A.2~~.2~~ 9

Condition A applies to the failure of a single channel in any RPS automatic trip Function. RPS coincidence logic is normally two-out-of-four.

If one RPS bistable trip unit or associated instrument channel is inoperable, startup or power operation is allowed to continue, providing the inoperable trip unit is placed in bypass or trip within 1 hour (Required Action A.1). With one channel in bypass, no additional random failure of a single channel could spuriously trip the reactor and a valid trip signal can still trip the reactor. With one channel in trip, an additional random failure of a single channel could spuriously trip the reactor. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to restore, bypass, or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel while ensuring that the risk involved in operating with the failed channel is acceptable.

Insert 1

~~The failed channel is restored to OPERABLE status or is placed in trip within [48] hours [or in accordance with the Risk Informed Completion Time Program] (Required Action A.2.1 or Required Action A.2.2). Required Action A.2.1 restores the full capability of the Function.~~

~~[Required Action A.2.2 places the Function in a one-out-of-three configuration. In this configuration, common cause failure of dependent channels cannot prevent trip.]~~

~~The Completion Time of [48] hours is based on operating experience, which has demonstrated that a random failure of a second channel occurring during the [48] hour period is a low probability event.~~ 9

B.1 ~~and B.2~~ 9

Condition B applies to the failure of two channels in any RPS automatic trip Function.

Required Action B.1 provides for placing one inoperable channel in bypass and the other channel in trip within the Completion Time of 1 hour. This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels while ensuring that the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed, the RPS is in a two-out-of-three logic; but with another channel failed, the RPS may be operating

9 1

INSERT 1

Additionally, the failed channel must be restored to OPERABLE status prior to entering MODE 2 following the next MODE 5 entry. The Completion Time of Required Action A.2 is based on adequate channel to channel independence, which allows a two-out-of-three channel operation since no single failure will cause or prevent a reactor trip. However, it is expected that the inoperable channel will be restored to OPERABLE status at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying restoration of the channel as well as any plant configuration changes required or shutting the plant down to repair the channel) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to repair the channel. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and NRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered even when the LCO is no longer applicable to ensure the channel is restored to OPERABLE prior to the next reactor startup from MODE 5 conditions.

BASES

ACTIONS (continued)

in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

Insert 2

~~One channel should be restored to OPERABLE status within [48] hours [or in accordance with the Risk Informed Completion Time Program] for reasons similar to those stated under Condition A. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 must be placed in trip if more than [48] hours have elapsed since the initial channel failure.~~ 9

C.1 and C.2

Local Power Density

The excore detectors are used to generate the internal ASI used as an input to the TM/LP and ~~APD~~ - High trips. Incore detectors provide a more accurate measurement of ASI. If one or more excore detectors cannot be calibrated to match incore detectors, power is restricted or reduced during subsequent operations because of increased uncertainty associated with using uncalibrated excore detectors. 3

The Completion Time of 24 hours is adequate to perform the SR while minimizing the risk of operating in an unsafe condition.

D.1, D.2.1, D.2.2.1, and D.2.2.2

Condition D applies to one automatic bypass removal channel inoperable. If the bypass removal channel for any operating bypass cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected RPS channel must be declared inoperable, as in Condition A, and the bypass either removed or the bypass removal channel repaired. The Bases for Required Actions and Completion Times are the same as discussed for Condition A.

E.1, E.2.1, and E.2.2

Condition E applies to two inoperable automatic bypass removal channels. If the bypass removal channels cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected

9 **INSERT 2** 1

One of the two inoperable channels will need to be restored to OPERABLE status prior to the next required CHANNEL FUNCTIONAL TEST, because channel surveillance testing on an OPERABLE channel requires that the OPERABLE channel be placed in bypass. However, it is not possible to bypass more than one RPS channel and placing a second channel in trip will result in a reactor trip. Therefore, if one RPS channel is in trip and a second channel is in bypass, a third inoperable channel would place the unit in LCO 3.0.3.

BASES

ACTIONS (continued)

RPS channels must be declared inoperable, as in Condition B, and the bypass either removed or the bypass removal channel repaired. Also, Required Action E.2.2 provides for the restoration of the one affected automatic trip channel to OPERABLE status within the rules of Completion Time specified under Condition B. Completion Times are consistent with Condition B.

F.1

Condition F is entered when the Required Action and associated Completion Time of Conditions A, B, C, D, or E are not met for the ~~Axial Power Distribution~~ Local Power Density and Loss of Load Trip Functions. 1

If the Required Actions associated with these Conditions cannot be completed within the required Completion Times, the reactor must be brought to a MODE in which the Required Actions do not apply. The allowed Completion Time of 6 hours to reduce THERMAL POWER to < 15% RTP is reasonable, based on operating experience, to decrease power to < 15% RTP from full power conditions in an orderly manner and without challenging plant systems.

G.1

Condition G is entered when the Required Action and associated Completion Time of Conditions A, B, C, D, E, or F are not met.

If the Required Actions associated with these Conditions cannot be completed within the required Completion Times, the reactor must be brought to a MODE in which the Required Actions do not apply. The allowed Completion Time of 6 hours to be in MODE 3 is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for any particular RPS Function are found in the SR column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and response time testing.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~In order for a plant to take credit for topical reports as the basis for justifying Frequencies, topical reports must be supported by an NRC staff SER that establishes the acceptability of each topical report for that plant (Ref. 9).~~
~~---~~

~~----------REVIEWER'S NOTE----------~~
~~Notes a and b are applied to the setpoint verification Surveillances for each RPS Instrumentation – Operating (Analog) Function in Table 3.3.1-1 unless one or more of the following exclusions apply:~~

~~1. Manual actuation circuits, automatic actuation logic circuits or instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.~~

~~2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.~~

~~3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.~~
~~---~~ 3

SR 3.3.1.1

Performance of the CHANNEL CHECK ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

~~[The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

SR 3.3.1.2

15 A daily calibration (heat balance) is performed when THERMAL POWER is ≥ ~~20~~%. The daily calibration shall consist of adjusting the "nuclear power calibrate" potentiometers to agree with the calorimetric calculation ~~if the absolute difference is > 1.5%~~. The "ΔT power calibrate" potentiometers are then used to null the "nuclear power - ΔT power" indicators on the RPS Reactor Power Calibration and Indication panel. Performance of the daily calibration ensures that the two inputs to the Q power measurement are indicating accurately with respect to the much more accurate secondary calorimetric calculation. 1 1

~~[The Frequency of 24 hours is based on plant operating experience and takes into account indications and alarms located in the control room to detect deviations in channel outputs.~~ 3

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~[REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 3

15 The Frequency is modified by a Note indicating this Surveillance must be performed within 12 hours after THERMAL POWER is ≥ ~~20~~% RTP. The secondary calorimetric is inaccurate at lower power levels. The 12 hours allows time requirements for plant stabilization, data taking, and instrument calibration. 1

A second Note indicates the ~~daily~~ calibration may be suspended during PHYSICS TESTS. This ensures that calibration is proper preceding and following physics testing at each plateau, recognizing that during testing, changes in power distribution and RCS temperature may render the calorimetric inaccurate. | R1

SR 3.3.1.3

Local Power Density It is necessary to calibrate the excore power range channel upper and lower subchannel amplifiers such that the internal ASI used in the TM/LP and ~~APD~~ - High trips reflects the true core power distribution as determined by the incore detectors. A Note to the Frequency indicates the Surveillance is required within 12 hours after THERMAL POWER is ≥ ~~[20]~~% RTP. Uncertainties in the excore and incore measurement process make it impractical to calibrate when THERMAL POWER is < ~~[20]~~% RTP. The Completion Time of 12 hours allows time for plant stabilization, data taking, and instrument calibration. If the excore detectors are not properly calibrated to agree with the incore detectors, power is restricted during subsequent operations because of increased uncertainty associated with using uncalibrated excore detectors. ~~[The 31 day Frequency is adequate, based on operating experience of the excore linear amplifiers and the slow burnup of the detectors. The excore readings are a strong function of the power produced in the peripheral fuel bundles and do not represent an integrated reading across the core. Slow changes in neutron flux during the fuel cycle can also be detected at this Frequency.~~ 1 15 15 3

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed on each RPS instrument channel, except Loss of Load and Power Rate of Change to ensure the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

In addition to power supply tests, The RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference ~~8~~ 9. These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. They include:

Bistable Tests

The bistable setpoint must be found to trip conservative with respect to the Allowable Values specified in the LCO and left set consistent with the assumptions of the plant specific setpoint analysis (Ref. ~~7~~). As-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference 10.

acceptance criteria

A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified ~~tolerance~~ around the setpoint. This is done with the affected RPS channel trip channel bypassed. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value (e.g., outside the conservative side of the as-found acceptance criteria band). 5 Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[LTSP]~~. Where a setpoint more conservative than the ~~[LTSP]~~ is used in the plant surveillance procedures ~~[NTSP]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and ~~/or~~ Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[LTSP]~~, then the channel shall be declared inoperable.

5 3 5 1 3 5

The second Note also requires that ~~[LTSP]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~. the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively

3 5

Matrix Logic Tests

Matrix Logic tests are addressed in LCO 3.3.~~3~~ 2. This test is performed one matrix at a time. It verifies that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. This test will detect any short circuits around the bistable contacts in the coincidence logic, such as may be caused by faulty bistable relay or trip channel bypass contacts. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

Trip Path Tests

Trip Path (Initiation Logic) tests are addressed in LCO 3.3.~~3~~. These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, opening the affected set of RTCBs. The RTCBs must then be closed prior to testing the other three initiation circuits, or a reactor trip may result.

~~[The Frequency of [92] days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 10).~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

SR 3.3.1.5

A CHANNEL CALIBRATION of the excore power range channels ensures that the channels are reading accurately and within ~~tolerance~~. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference ~~[~~10~~]~~.

A Note is added stating that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by

BASES

SURVEILLANCE REQUIREMENTS (continued)

performing the ~~daily~~ calorimetric calibration (SR 3.3.1.2) and the ~~monthly~~ linear subchannel gain check (SR 3.3.1.3). In addition, associated control room indications are continuously monitored by the operators. 1

~~[The Frequency of 92 days is acceptable, based on plant operating experience, and takes into account indications and alarms available to the operator in the control room.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 3

SR 3.3.1.5 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value. 5
Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition.
The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[LTSP]~~. Where a setpoint more conservative than the ~~[LTSP]~~ is used in the plant surveillance procedures ~~[NTSP]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[LTSP]~~, then the channel shall be declared inoperable. 5 5 3 5

acceptance criteria band
acceptance criteria band
NTSP
acceptance criteria bands
acceptance criteria band
NTSP

BASES

SURVEILLANCE REQUIREMENTS (continued)

The second Note also requires that ~~[LTSP]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~.

NTSP

acceptance criteria bands

3 5 R1

the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively

SR 3.3.1.6

A CHANNEL FUNCTIONAL TEST on the Loss of Load and Power Rate of Change channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. ~~The Loss of Load pressure sensor cannot be tested during reactor operation without closing the high pressure TSV, which would result in a turbine trip or reactor trip.~~ The Power Rate of Change - High trip Function is required during startup operation and is bypassed when shut down or > 15% RTP.

The Loss of Load trip Function is not tested during reactor operation because performing this test during operation could cause a pressure perturbation in the emergency trip fluid header common to all four channels resulting in a turbine trip and reactor trip.

1 R1

SR 3.3.1.6 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition.

acceptance criteria band

5

The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[LTSP]~~. Where a setpoint more conservative than the ~~[LTSP]~~ is used in the plant surveillance procedures ~~[NTSP]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[LTSP]~~, then the channel shall be declared inoperable.

acceptance criteria band

NTSP

5

acceptance criteria bands

3 5

acceptance criteria band

NTSP

5

BASES

SURVEILLANCE REQUIREMENTS (continued)

NTSP

acceptance criteria bands

The second Note also requires that ~~[LTSP]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~.

3 5

the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively

SR 3.3.1.7

SR 3.3.1.7 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.1.4, except SR 3.3.1.7 is applicable only to bypass Functions and is performed once within 92 days prior to each startup. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Proper operation of bypass permissives is critical during plant startup because the bypasses must be in place to allow startup operation and must be removed at the appropriate points during power ascent to enable certain reactor trips. Consequently, the appropriate time to verify bypass removal function OPERABILITY is just prior to startup. The allowance to conduct this test within 92 days of startup is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 10). Once the operating bypasses are removed, the bypasses must not fail in such a way that the associated trip Function gets inadvertently bypassed. This feature is verified by the trip Function CHANNEL FUNCTIONAL TEST, SR 3.3.1.4. Therefore, further testing of the bypass function after startup is unnecessary.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a CHANNEL CALIBRATION.

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference ~~[~~10~~]~~. 3

SR 3.3.1.8 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value (e.g., outside the conservative side of the as-found acceptance criteria band) 5 Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition.

The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[LTSP]~~. Where a setpoint more conservative than the ~~[LTSP]~~ is used in the plant surveillance procedures ~~[NTSP]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[LTSP]~~, then the channel shall be declared inoperable. 5 5 3 5

The second Note also requires that ~~[LTSP]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~ . the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively 5 3

~~[The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4

The Surveillance is modified by a Note to indicate that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the calorimetric calibration (SR 3.3.1.2) and the linear subchannel gain check (SR 3.3.1.3).

SR 3.3.1.9

This SR ensures that the RPS RESPONSE TIMES are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the RTCBs open. ~~[Response times are conducted on an [18] month STAGGERED TEST BASIS. This results in the interval between successive surveillances of a given channel of n x 18 months, where n is the number of channels in the function. The Frequency of [18] months is based upon operating experience, which has shown that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Also, response times cannot be determined at power, since equipment operation is required.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

Testing may be performed in one measurement or in overlapping segments, with verification that all components are tested.

~~----------REVIEWER'S NOTE----------~~
~~Applicable portions of the following TS Bases are applicable to plants adopting CEOG Topical Report CE NPSD-1167-1, "Elimination of Pressure Sensor Response Time Testing Requirements," and the methodology contained in Attachment 1 to TSTF-569.~~
~~----------~~ 4

Response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements," (Ref. 11) provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the Topical Report. The response time may be verified for components that replace the components that were previously evaluated in Ref. 11 provided that the components have been evaluated in accordance with the NRC approved methodology as discussed in Attachment 1 to TSTF-569, "Methodology to Eliminate Pressure Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing," (Ref. 12). Response time verification for other sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and reverified after maintenance that may adversely affect the sensor response time.

A Note is added to indicate that the neutron detectors are excluded from RPS RESPONSE TIME testing because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.2).

REFERENCES

1. Technical Requirements Manual (TRM)

2 ~~1~~. Regulatory Guide 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation."

3 ~~2~~. 10 CFR 50, Appendix A, GDC 21.

4 ~~3~~. 10 CFR ~~100~~ 50.67.

5 ~~4~~. IEEE Standard 279-1971, April 5, 1972. 1

BASES

REFERENCES (continued)

~~5~~. FSAR, Chapter ~~[14]~~. U 15

~~6~~. 10 CFR 50.49.

IC-3.17, “Instrument Setpoint Methodology for Nuclear Power Plants”

~~7. "Plant Protection System Selection of Trip Setpoint Values."~~

~~8~~. FSAR, Section ~~[~~7.2~~]~~. U

~~9. NRC Safety Evaluation Report, [Date].~~

10. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989.

11. CEOG Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements."

12. Attachment 1 to TSTF-569, "Methodology to Eliminate Pressure Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing."

BASES

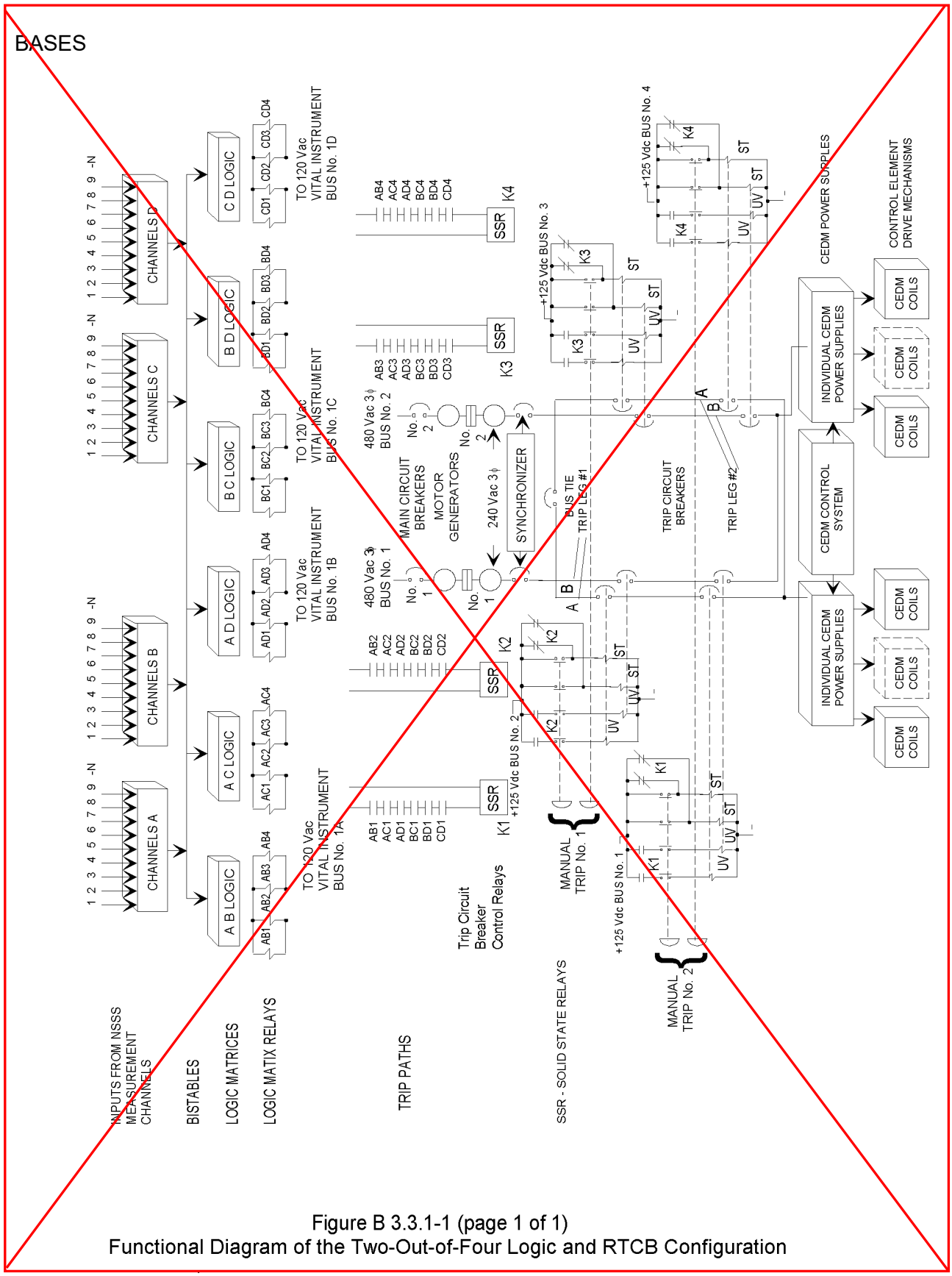

Figure B 3.3.1-1 (page 1 of 1)
Functional Diagram of the Two-Out-of-Four Logic and RTCB Configuration

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.1 BASES, REACTOR PROTECTIVE SYSTEM (RPS) INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The heading for ISTS 3.3.3 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. PSL does not use the terms “as-found tolerance” and “Limiting Trip Setpoint (LTSP)” in plant specific instrument calculations or the Updated Final Safety Analysis Report. Therefore, the use of these terms in Footnotes (a) and (b) in ISTS Table 3.3.1-1 are revised in the ITS to include the plant specific terminology. To comply with the guidance provided in NRC Regulatory Issue Summary 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36, ‘Technical Specifications,’ Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels,” the PSL setpoint calculations have been structured to include a Nominal Trip Setpoint (NTSP), also known as the field trip setpoint, and determination of an OPERABILITY limit range, referred to in CTS as acceptance criteria band. For PSL, the Nominal Trip Setpoint (NTSP) is synonymous with the Limiting Trip Setpoint (LTSP) and the as-found acceptance criteria band (i.e., OPERABILITY limit range) is synonymous with the as-found tolerance.

6. Figure B 3.3.1-1 is removed because the figure is included in the UFSAR.

7. Editorial clarification.

8. Changes made to reflect changes made to the Specification consistent with the current licensing basis as approved in License Amendment 243 (Unit 1) and Amendment 194 (Unit 2), dated October 31, 2017 (NRC ADAMS Accession Number ML17257A015).

9. Changes made to reflect changes made to the Specification consistent with the current licensing basis with Bases of ACTIONS A and B adapted from the Bases ISTS Specification 3.3.1 (Digital), and the Bases of Specification SR 3.0.3.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.1, REACTOR PROTECTIVE SYSTEM (RPS) INSTRUMENTATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

3.3.2, Reactor Protective System (RPS) Logic and Trip Initiation

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3/4.3 INSTRUMENTATION

SYSTEM (RPS) LOGIC AND TRIP INITIATION

3/4.3.1 REACTOR PROTECTIVE ~~INSTRUMENTATION~~

LIMITING CONDITION FOR OPERATION

LCO 3.3.2

3.3.1.1 ~~As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1~~ shall be OPERABLE.

A02

Six channels of RPS Matrix Logic, four channels of RPS Initiation Logic, four channels of reactor trip circuit breakers (RTCB)s, and four channels of Manual Trip

M01

Applicability

APPLICABILITY: ~~As shown in Table 3.3-1.~~

MODES 1 and 2,
MODES 3, 4, and 5, with any RTCBs closed and any control element assemblies capable of being withdrawn.

A03

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1
SR 3.3.2.2
SR 3.3.2.3

4.3.1.1.1 Each ~~reactor protective instrumentation~~ channel shall be demonstrated OPERABLE by the performance of the ~~CHANNEL CHECK, CHANNEL CALIBRATION and~~ CHANNEL FUNCTIONAL TEST ~~operations during the modes and at the frequencies shown in Table 4.3-1~~.

A01

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing. Each test shall include at least one channel per function.

See ITS 3.3.1

Add proposed SR 3.3.2.4

M02

TABLE 3.3-1

SYSTEM (RPS) LOGIC AND TRIP INITIATION

REACTOR PROTECTIVE ~~INSTRUMENTATION~~

MODES 3, 4, and 5, with any RTCBs closed and any control element assemblies capable of being withdrawn

A03

LA01

FUNCTIONAL ~~UNIT~~	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	~~2~~	~~1~~	~~2~~	1, 2 and ~~*~~	1
2. Power Level – High	4	2(a)	3(f)	1, 2	2
3. Reactor Coolant Flow – Low	4/SG	2(a)/SG	3/SG	1, 2 (e)	2
4. Pressurizer Pressure – High	4	2	3	1, 2	2
5. Containment Pressure – High	4	2	3	1, 2	2
6. Steam Generator Pressure – Low	4/SG	2(b)/SG	3/SG	1, 2	2
7. Steam Generator Water Level – Low	4/SG	2/SG	3/SG	1, 2	2
8. Local Power Density – High	4	2(c)	3	1	2
9. Thermal Margin/Low Pressure	4	2(a)	3	1, 2 (e)	2
9a. Steam Generator Pressure Difference – High	4	2(a)	3	1, 2 (e)	2
10. Loss of Turbine – Hydraulic Fluid Pressure - Low	4	2(c)	3	1	2

4

4

LCO 3.3.2 Applicability ACTION A

A02

See ITS 3.3.1

TABLE 3.3-1 (Continued)

SYSTEM (RPS) LOGIC AND TRIP INITIATION

REACTOR PROTECTIVE ~~INSTRUMENTATION~~

LA01

FUNCTIONAL ~~UNIT~~	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
11. Wide Range Logarithmic Neutron Flux Monitor						
a. Startup and Operating -- Rate of Change of Power – High	4	2(d)	3	1**, 2 and *	2	See ITS 3.3.1
b. Shutdown	4	0	2	3, 4, 5	3	See ITS 3.3.11
12. ~~Reactor Protection System~~ Logic (RPS initiation logic)	~~4~~	~~2~~	4	1, 2~~*~~	4	A03
13. Reactor Trip Breakers (Circuit, RTCBs)	~~4~~	~~2~~	4	1, 2~~*~~	4	

LCO 3.3.2 Applicability ACTION B

LCO 3.3.2 Applicability ACTION B

MODES 3, 4, and 5, with any RTCBs closed and any control element assemblies capable of being withdrawn

TABLE 3.3-1 (Continued)

TABLE NOTATION

Applicability * With ~~the protective system trip breakers in the~~ closed ~~position~~ and ~~the~~ CEA ~~drive system~~ capable of ~~CEA~~ withdrawal.

any RTCBs

any

being

A03

** Mode 1 applicable only when Power Range Neutron Flux power ≤ 15% of RATED THERMAL POWER.

(a) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is ≥ 1% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is ≥ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is ≥ 10^{-4}% and Power Range Neutron Flux power ≤ 15% of RATED THERMAL POWER.

(e) Deleted.

(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

See ITS 3.3.1

ACTION STATEMENTS

One manual trip channel inoperable in MODE 1 or 2

ACTION A (Manual Trip)

ACTION 1 - ~~With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement,~~ restore ~~the inoperable channel~~ to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and ~~/or~~ open the protective system trip breakers.

ACTION E (Manual Trip)

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

See ITS 3.3.1

Add proposed ACTION A (Matrix Logic)

M01

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

b. Within one hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on than channel provided the other inoperable channel is placed in the tripped condition.

See ITS 3.3.1

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

See ITS 3.3.11

One channel RTCB or initiation logic inoperable in MODE 1 or 2.

ACTION B (RTCBs and Initiation Logic)

ACTION 4 - ~~With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1~~.

L01

LA02

open the affected RTCBs within 1 hour

Add proposed ACTION C

M03

Add proposed ACTION D

M03

Add proposed ACTION E (for Initiation Logic, RTCBs and second condition)

M04

ITS

A01

ITS 3.3.2

TABLE 4.3-1

REACTOR PROTECTIVE ~~INSTRUMENTATION~~ SURVEILLANCE REQUIREMENTS

SYSTEM (RPS) LOGIC AND TRIP INITIATION

	FUNCTIONAL ~~UNIT~~	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
SR 3.3.2.3	1. Manual Reactor Trip	N/A	N.A.	S/U(1)	N/A
	2. Power Level – High				
	a. Nuclear Power	SFCP	SFCP(2), SFCP(3), SFCP(5)	SFCP	1,2
	b. ΔT Power	SFCP	SFCP(4), SFCP	SFCP	1
	3. Reactor Coolant Flow – Low	SFCP	SFCP	SFCP	1, 2
	4. Pressurizer Pressure – High	SFCP	SFCP	SFCP	1, 2
	5. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2
	6. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2
	7. Steam Generator Water Level – Low	SFCP	SFCP	SFCP(6, 7)	1, 2
	8. Local Power Density – High	SFCP	SFCP	SFCP	1
	9. Thermal Margin/Low Pressure	SFCP	SFCP	SFCP	1, 2
	9a. Steam Generator Pressure Difference – High	SFCP	SFCP	SFCP	1, 2
	10. Loss of Turbine -- Hydraulic Fluid Pressure – Low	N.A.	N.A.	S/U(1)	N.A.
	11. Wide Range Logarithmic Neutron Flux Monitor	SFCP	N.A.	S/U(1)	1, 2, 3, 4, 5 and *
SR 3.3.2.2 Applicability	12. ~~Reactor Protection System~~ Logic (RPS initiation logic)	N.A.	N.A.	SFCP ~~and S/U(1)~~	1, 2 and *
SR 3.3.2.1 Applicability	13. Reactor Trip Breakers (Circuit) (RTCBs)	N.A.	N.A.	SFCP	1, 2 and *

See ITS 3.3.1

L02

TABLE 4.3-1 (Continued)

TABLE NOTATION

Applicability * - With reactor trip breaker closed.

~~(1) - If not performed in previous 7 days.~~

L02

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Pwr – ΔT Pwr." During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to ≤ 90% of the maximum allowed THERMAL POWER level with the existing Reactor Coolant Pump combination.

(4) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

(5) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(6) - If the as-found setpoint is either outside its predefined as-found acceptance criteria band or is not conservative with respect to the Allowable Value, then the channel shall be declared inoperable and shall be evaluated to verify that it is functioning as required before returning the channel to service.

(7) - The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Field Trip Setpoint, otherwise that channel shall not be returned to OPERABLE status. The Field Trip Setpoint and the methodology used to determine the Field Trip Setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in the UFSAR Section 7.2.

See ITS 3.3.1

3/4.3 INSTRUMENTATION

SYSTEM (RPS) LOGIC AND TRIP INITIATION

3/4.3.1 REACTOR PROTECTIVE ~~INSTRUMENTATION~~

LIMITING CONDITION FOR OPERATION

LCO 3.3.2

3.3.1 ~~As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1~~ shall be OPERABLE.

Six channels of RPS Matrix Logic, four channels of RPS Initiation Logic, four channels of reactor trip circuit breakers (RTCB)s, and four channels of Manual Trip

M01

Applicability

APPLICABILITY: ~~As shown in Table 3.3-1.~~

MODES 1 and 2,
MODES 3, 4, and 5, with any RTCBs closed and any control element assemblies capable of being withdrawn.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1
SR 3.3.2.2
SR 3.3.2.3

4.3.1.1 Each ~~reactor protective instrumentation~~ channel shall be demonstrated OPERABLE by the performance of the ~~CHANNEL CHECK, CHANNEL CALIBRATION and~~ CHANNEL FUNCTIONAL TEST ~~operations for the MODES and at the frequencies shown in Table 4.3-1~~.

A01

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing. Each test shall include at least one channel per function.

See ITS 3.3.1

TABLE 3.3-1

REACTOR PROTECTIVE ~~INSTRUMENTATION~~

SYSTEM (RPS) LOGIC AND TRIP INITIATION

LA01

	FUNCTION~~AL UNIT~~	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
ACTION A ACTION C ACTION D	1. Manual Reactor Trip	4 4	~~2~~ ~~2~~	4 4	1, 2 3*, 4*, 5*	1 5	
	2. Variable Power Level – High	4	2(a)(d)	3	1, 2	2	
	3. Pressurizer Pressure – High	4	2	3	1, 2	2	
	4. Thermal Margin/Low Pressure	4	2(a)(d)	3	1, 2	2	
	5. Containment Pressure – High	4	2	3	1, 2	2	
	6. Steam Generator Pressure – Low	4/SG	2/SG(b)	3/SG	1, 2	2	See ITS 3.3.1
	7. Steam Generator Pressure Difference – High	4	2(a)(d)	3	1, 2	2	
	8. Steam Generator Level – Low	4/SG	2/SG	3/SG	1, 2	2	
	9. Local Power Density – High	4	2(c)(d)	3	1	2	
	10. Loss of Component Cooling Water to Reactor Coolant Pumps	4	2	3	1, 2	2	
ACTION B ACTION C	11. ~~Reactor Protection System~~ Logic (RPS initiation logic)	~~4~~	~~2~~	~~3~~ 4	1, 2 3*, 4*, 5*	2 5	A04
ACTION B ACTION C	12. Reactor Trip Breakers (Circuit, RTCBs)	~~4~~	~~2(f)~~	4	1, 2 3*, 4*, 5*	4 5	
	13. Wide Range Logarithmic Neutron Flux Monitor a. Startup and Operating – Rate of Change of Power – High	4	2(e)(g)	3	1**, 2	2	See ITS 3.3.1
	b. Shutdown	4	0	2	3, 4, 5	3	See ITS 3.3.11
	14. Reactor Coolant Flow – Low	4/SG	2/SG(a)(d)	3/SG	1, 2	2	
	15. Loss of Load (Turbine Hydraulic Fluid Pressure – Low)	4	2(c)	3	1	2	See ITS 3.3.1

TABLE 3.3-1 (Continued)

TABLE NOTATION

Applicability * With ~~the protective system trip breakers in the~~ closed ~~position, the~~ CEA ~~drive system~~ capable of ~~CEA~~ withdrawal~~, and fuel in the reactor vessel~~.

any RTCBs

and any

being

A05

See ITS 3.3.1

** Mode 1 applicable only when Power Range Neutron Flux power ≤ 15% of RATED THERMAL POWER.

(a) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER in conjunction with (d) below; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is greater than or equal to 15% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq$ 10^{-4}% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

(f) ~~Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.~~

LA03

(g) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

See ITS 3.3.1

ACTION STATEMENTS

One manual trip channel inoperable in MODE 1 or 2

ACTION A (Manual Trip)

ACTION E (Manual Trip)

ACTION 1 - ~~With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status~~ within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and~~/or~~ open the protective system trip breakers.

Add proposed ACTION A (Matrix Logic)

M01

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

One channel of Initiation Logic inoperable in MODE 1 or 2.

ACTION 2 - a. ~~With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition~~ within 1 hour. ~~The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.~~

ACTION B (Initiation Logic)

LA04

open the affected RTCBs

b. ~~With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:~~

Two channels of Initiation Logic affecting the same trip leg inoperable in MODE 1 or 2.

ACTION D (Initiation Logic Modes 1 and 2 only)

1. ~~Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.~~

M05

ACTION D (Initiation Logic Modes 1 and 2 only)

2. ~~All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition.~~

Open the affected RTCBs immediately.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed
1. Safety Channel – Nuclear Instrumentation	
Wide Range	Rate of Change of Power – High (RPS)
Linear Range	Variable Power Level – High (RPS) Local Power Density – High (RPS) Thermal Margin/Low Pressure (RPS)
2. Pressurizer Pressure -	Pressurizer Pressure – High (RPS) Thermal Margin/Low Pressure (RPS) Pressurizer Pressure – Low (ESF)
3. Containment Pressure -	Containment Pressure – High (RPS) Containment Pressure – High (ESF)
4. Steam Generator Pressure -	Steam Generator Pressure – Low (RPS) Thermal Margin/Low Pressure (RPS) AFAS-1 and AFAS-2 (AFAS) Steam Generator Pressure – Low (ESF)
5. Steam Generator Level -	Steam Generator Level – Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)

See ITS 3.3.1

Add proposed ACTION D (for Manual Trip, RTCBs; and Initiation Logic in Modes 3, 4 and 5)

M03

Add proposed ACTION E (for Initiation Logic and second Condition)

M04

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

See ITS 3.3.1

ACTION 2 - (Continued)

6.	Cold Leg Temperature	Variable Power Level – High (RPS) Thermal Margin/Low Pressure (RPS) Local Power Density – High (RPS)
7.	Hot Leg Temperature	Variable Power Level – High (RPS) Thermal Margin /Low Pressure (RPS) Local Power Density – High (RPS)

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes*. Verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

See ITS 3.3.11

One channel of RTCBs inoperable in MODE 1 or 2, open the affected RTCBs

ACTION B (RTCBs)

ACTION E (RTCBs)

ACTION 4 - ~~With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition~~ within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; ~~however, one channel may be bypassed for up to 1 hour, provided the trip breakers of any inoperable channel are in the tripped condition, for surveillance testing per Specification 4.3.1.1.~~

LA02

One channel of Manual Trip, RTCBs, or Initiation Logic inoperable in MODE 3, 4, or 5

ACTION C

ACTION 5 - ~~With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status~~ within 48 hours ~~or~~ open the ~~reactor trip breakers within the next hour~~.

affected RTCBs

M06

* Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

See ITS 3.3.11

TABLE 4.3-1

SYSTEM (RPS) LOGIC AND TRIP INITIATION

REACTOR PROTECTIVE ~~INSTRUMENTATION~~ SURVEILLANCE REQUIREMENTS

	FUNCTIONAL ~~UNIT~~	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
SR 3.3.2.3	1. Manual Reactor Trip	N/A	N.A.	S/U(1)	1, 2, 3*, 4*, 5*	
	2. Variable Power Level – High					
	a. Nuclear Power	SFCP	SFCP(2), SFCP(3), SFCP(4)	SFCP	1,2	
	b. ΔT Power	SFCP	SFCP(5), SFCP(4)		1	
	3. Pressurizer Pressure – High	SFCP	SFCP	SFCP	1, 2	
	4. Thermal Margin/Low Pressure	SFCP	SFCP	SFCP	1, 2	
	5. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2	See ITS 3.3.1
	6. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2	
	7. Steam Generator Pressure Difference – High	SFCP	SFCP	SFCP	1, 2	
	8. Steam Generator Level – Low	SFCP	SFCP	SFCP(8, 9)	1, 2	
	9. Local Power Density – High	SFCP	SFCP	SFCP	1	
	10. Loss of Component Cooling Water to Reactor Coolant Pumps	N.A.	N.A.	SFCP	N.A.	
SR 3.3.2.2	11. ~~Reactor Protection System~~ Logic (RPS initiation logic)	N.A.	N.A.	SFCP~~(7)~~	1, 2, 3*, 4*, 5*	LA05

TABLE 4.3-1 (Continued)

SYSTEM (RPS) LOGIC AND TRIP INITIATION

REACTOR PROTECTIVE ~~INSTRUMENTATION~~ SURVEILLANCE REQUIREMENTS

	FUNCTION~~AL UNIT~~	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
SR 3.3.2.1 SR 3.3.2.4	12. Reactor Trip Breakers (Circuit, RTCBs)	N.A.	N.A.	~~S/U(1),~~ SFCP, SFCP(6)	1, 2, 3*, 4*, 5*	L02
	13. Wide Range Logarithmic Neutron Flux Monitor	SFCP	SFCP	S/U(1),SFCP	1, 2, 3, 4, 5	
	14. Reactor Coolant Flow – Low	SFCP	SFCP	SFCP	1, 2	See ITS 3.3.1
	15. Loss of Load (Turbine Hydraulic Fluid Pressure – Low)	SFCP	N.A.	SFCP	1	

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - Only if the reactor trip breakers are in the closed position and the CEA drive system is capable of CEA withdrawal.

~~(1) - Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.~~

L02

See ITS 3.3.1

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Power – ΔT Power". During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to ≤ 90% of the maximum allowed THERMAL POWER level with the existing reactor coolant pump combination.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

SR 3.3.2.4

(6) - In accordance with the Surveillance Frequency Control Program and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include verification of the independent OPERABILITY of the undervoltage and shunt trips.

~~(7) - The fuse circuitry in the matrix fault protection circuitry shall be determined to be OPERABLE by testing with the installed test circuitry.~~

LA05

(8) - If the as-found channel setpoint is either outside its predefined as-found acceptance criteria band or is not conservative with respect to the Allowable Value, then the channel shall be declared inoperable and shall be evaluated to verify that it is functioning as required before returning the channel to service.

(9) - The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Field Trip Setpoint, otherwise that channel shall not be returned to OPERABLE status. The Field Trip Setpoint and the methodology used to determine the Field Trip Setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in UFSAR Section 7.2.

See ITS 3.3.1

DISCUSSION OF CHANGES
ITS 3.3.2, REACTOR PROTECTIVE SYSTEM (RPS) LOGIC AND INITIATION INSTRUMENTATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 1 only:** CTS Table 3.3-1 Functional Unit 1 provides requirements for Manual Reactor Trip. The Total No. Of Channels and Minimum Channels Operable columns specify 2 channels. ITS LCO 3.3.2, in part, requires four channels of Manual Trip to be OPERABLE. This changes the CTS by requiring four Manual Reactor Trip channels instead of two channels.

The purpose of the required number of channels is to ensure a manual reactor trip can be initiated coincident with a single failure. As shown in UFSAR Figure 7.2-1, the Reactor Protective System manual trip function is comprised of two manual trip push buttons per Manual Trip circuit; Manual Trip #1 and #2. Two pushbuttons associated with either Manual Trip circuit are required to trip the reactor. CTS Table 3.3-1 Functional Unit 1 specifies a Manual Trip Circuit as a channel (i.e., Manual Trip #1 and #2). ISTS LCO 3.3.3 (ITS LCO 3.3.2) and the associated ACTIONS and Surveillances are formatted to address each individual manual trip push button as a channel. The PSL Unit 1 and Unit 2 manual reactor trip circuit design is consistent with the manual reactor trip circuit design discussed in the Bases of ISTS 3.3.3. The Background section of the ISTS 3.3.3 Bases states, in part, “Each set of RTCBs is operated by… a Manual Trip push button…. There are four Manual Trip push buttons, arranged in two sets of two…. Depressing both push buttons in either set will result in a reactor trip.” Therefore, changing the minimum number of required channels from 2 to 4 results is an equivalent requirement; two sets of two manual push buttons must be OPERABLE and one set of two manual push buttons are required to initiate a manual reactor trip.

This change is a presentation preference and is consistent with NUREG-1432, Rev. 5 and PSL Unit 2 CTS Table 3.3-1 Functional Unit 1 Total No. of Channels and Minimum Channels Operable columns. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 **Unit 1 only**: CTS Table 3.3-1 lists the applicability of the RPS Manual Reactor Trip (Functional Unit 1) as MODES 1, 2 and * CTS Table 3.3-1 also lists the applicability of the Reactor Protective System Logic (Functional Unit 12) and Reactor Trip Breakers (Functional Unit 13) as MODES 1 and 2*. Footnote * provides the clarification that the applicability includes “with the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.” ITS 3.3.2 Applicability states MODES 1 and 2, and MODES 3, 4, and 5, with any RTCBs closed and any control element assemblies capable of

being withdrawn. This changes the CTS by specifically adding MODES 3, 4, and 5 to the specific condition of with any RTCBs closed and any control element assemblies capable of being withdrawn

The purpose of the CTS Applicability is to ensure the RPS manual trip channel and logic and initiation instrumentation is also applicable when the plant is shutdown if the RTCBs are closed and the CEAs are capable of being withdrawn to ensure a withdrawn CEA can be inserted if required. Clarifying the Applicability of MODES 3, 4, and 5 avoids any confusion of the RPS requirements when the reactor vessel head is de-tensioned or removed (i.e., MODE 6). Other Technical Specifications (e.g., LCO 3.9.1, “Boron Concentration,”) ensure adequate Shutdown Margin is provided in MODE 6. This change is consistent with NUREG-1432, Rev. 5 and the Applicability for the same RPS Functions in Unit 2 CTS Table 3.3-1. This change is solely a clarification of the existing requirement and is designated as administrative because it does not result in technical changes to the CTS.

A04 **Unit 2 only**: Unit 2 CTS Table 3.3-1 Function 11 (Reactor Protection System Logic) requires 3 channels in the Minimum Channels Operable column for MODES 1 and 2. ITS 3.3.2 requires four channels for Function 11 (RPS Initiation Logic) to be OPERABLE for MODES 1 and 2. This changes the Unit 2 CTS to require four Reactor Protective System Logic (Function 11) channels instead of three channels.

The purpose of the required number of channels is to ensure the RPS Initiation Logic trip can be initiated coincident with a single failure. As shown in UFSAR Figure 7.2-1, the Reactor Protective System RPS Initiation Logic function is comprised of two channels per trip circuit. Changing the minimum number of required channels from 3 to 4 results in an equivalent requirement; two channels per trip circuit are required to satisfy single failure criteria.

This change is consistent with NUREG-1432, Rev. 5 and PSL Unit 2 CTS Table 3.3-1 Functional Unit 11 Total No. of Channels, Minimum Channels Operable, and Channels to Trip columns. This change is designated as administrative because it does not result in technical changes to the CTS.

A05 **Unit 2 only:** CTS 3.3.1 Table 3.3-1 Footnote* states, in part. “and fuel in the reactor vessel.” ITS 3.3.2 does not include this detail related to an operational MODE. The ITS definition of a MODE states that a MODE, in part, with fuel in the reactor vessel. This changes the CTS by removing redundant information already addressed by the Technical Specification definitions.

The purpose of the statement regarding fuel in the reactor in CTS Table 3.3-1 Footnote * is to clarify that the Applicability applies with fuel in the reactor vessel. The equivalent ITS 3.3.2 Applicability clarifies the requirements are applicable in MODES 3, 4, and 5, which include fuel in the reactor vessel. The ITS definition of a MODE is defined, in part, with fuel in the reactor vessel. ITS 3.3.2 retains the requirement for the RPS Logic and Trip Initiation to be OPERABLE in MODES 3, 4, and 5 with any RTCBs closed and any control element assemblies capable of being withdrawn. This change is solely a presentation preference difference and

is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.3.1.1 and Unit 2 CTS 3.3.1 do not specifically identify the RPS Matrix Logic channels as a functional unit required to satisfy RPS logic OPERABILITY. ITS 3.3.2 requires six channels of RPS Matrix Logic to be OPERABLE. This changes CTS by specifically adding six channels of RPS Matrix Logic to the channel OPERABILITY requirements and adding the associated proposed ACTION for inoperable RPS Matrix Logic channels.

The proposed change results in more specific requirements for the RPS Logic and Trip Initiation channels including the actions to be taken for inoperability. The proposed change is acceptable because the matrix logic is currently part of the RPS logic and trip initiation system but are not specifically identified within the OPERABILITY requirements. ITS 3.3.2 specifically identifies the six RPS Matrix Logic channels as an LCO requirement. The PSL Unit 1 and Unit 2 RPS matrix logic design is consistent with the RPS matrix logic design discussed in the Bases of ISTS 3.3.3. ITS 3.3.2 ACTION A requires the inoperable channels to be restored to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program when one Matrix Logic channel is inoperable or when three Matrix Logic channels are inoperable due to a common power source failure de-energizing three matrix power supplies. The Completion Time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second Matrix Logic channel is low during any given 48 hour interval. By providing specific requirements for the six RPS Matrix Logic channels, the proposed change provides additional assurance that the RPS function is available when required. This change conforms the PSL CTS to the ISTS. This change is considered a more restrictive change because additional specific requirements have been added that are not required by the CTS.

M02 **Unit 1 only**: CTS 3.3.1.1 does not have a specific surveillance requirement that requires an additional CHANNEL FUNCTIONAL TEST that includes a separate verification of the undervoltage and shunt trips on each RTCB channel. ITS SR 3.3.2.4 requires the performance of a CHANNEL FUNCTIONAL TEST, including separate verification of the undervoltage and shunt trips on each RTCB channel in accordance with the SFCP. This changes the CTS by adding an additional surveillance requirement ITS SR 3.3.2.4.

The purpose of SR 3.3.2.4 is to individually test all four sets of undervoltage coils and all four sets of shunt trip coils. This change is acceptable because it ensures that every undervoltage coil and every shunt trip coil is capable of performing its intended function and that no single failure of an RTCB component will prevent a reactor trip. PSL controls periodic Frequencies for Surveillances in accordance with the SFCP per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, the initial Frequency in accordance with the Surveillance Frequency Control Program will be specified as 18 months consistent with ISTS 3.3.3.4 and based on the

need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

The SFCP was established as described in FPL (PSL Unit 1 and Unit 2) "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program" (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066).

This change is acceptable and is designated as more restrictive since the CTS does not currently require this verification for the undervoltage and shunt trips on each RTCB channel.

M03 Unit 1 CTS 3.3.1.1 does not provide specific Actions for one or two inoperable channels of Manual Trip, RTCBs, or Initiation Logic. Unit 2 CTS 3.3.1 does not provide specific Actions for two channels of Manual Trip, RTCBs, or Initiation Logic affecting the same trip leg inoperable when in MODES 3, 4, and 5. ITS 3.3.2 ACTION C requires affected RTCBs to be opened within 48 hours with one channel of Manual Trip, RTCBs, or Initiation Logic inoperable in MODE 3, 4, or 5. ITS 3.3.2 ACTION D requires the affected RTCBs to be opened immediately with two channels of Manual Trip, RTCBs, or Initiation Logic affecting the same trip leg inoperable when in any required MODE of applicability. This changes Unit 1 CTS 3.3.1.1 by adding specific ACTIONs (ITS 3.3.2 ACTION C) for one channel inoperable in MODE 3, 4, or 5 and (ITS 3.3.2 ACTION D) for two channels of Manual Trip, RTCBs, or Initiation Logic affecting the same trip leg inoperable for all applicable MODES; and, changes Unit 2 CTS 3.3.1 by adding a specific ACTION (ITS 3.3.2 ACTION D) for two channels of Manual Trip, RTCBs, or Initiation Logic channel affecting the same trip leg inoperable for MODES 3, 4, and 5, with any RTCBs closed and any control element assemblies capable of being withdrawn.

The purpose of ITS 3.3.2 ACTION C is to remove the need for the affected channel by performing its associated safety function. The purpose of ITS 3.3.2 ACTION D is to allow for a single failure within a trip leg when one or both trip function channels in the same trip leg fails in a non-trip condition. This change is acceptable because with both trip channels of a trip function in the same trip leg inoperable opening of the affected RTCBs ensures the trip capability is maintained. This change is considered a more restrictive change because additional requirements have been added to the CTS.

M04 Unit 1 CTS 3.3.1.1 and Unit 2 CTS 3.3.1 do not provide specific Actions when the Action and Completion time for more than one Initiation Logic or RTCB channel inoperable. ITS ACTION E requires the unit to be placed in MODE 3 in 6 hours and all RTCBs opened in 6 hours when the Required Action and Associated Completion Time of Condition A (manual trip channel or matrix logic channel inoperable), Condition B (RTCB or Initiation Logic channel inoperable) or

Condition D (two channels affecting the same trip leg inoperable) are not met or "one or more Functions with two or more Manual Trip, Matrix Logic, Initiation Logic, or RTCB channels inoperable for reasons other than Condition A or D." This changes the CTS by adding Action E in its entirely for the Initiation Logic and RTCB functions and adding the second Condition for the manual trip function.

The purpose of ITS ACTION E is to place the unit in a MODE where the LCO does not apply and ensuring no CEA withdrawal occurs when Required Actions and associated Completion Times are not met or when one or more Functions have two or more channels inoperable for reasons other than Condition A or D. This change is acceptable because the ACTIONS specified in ITS 3.3.2 adopt ISTS for placing the unit outside the MODE of applicability. A Completion Time of 6 hours is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems and to open RTCBs. This change is considered a more restrictive change because additional requirements have been added to the CTS.

M05 **Unit 2 only:** CTS 3.3.1 Table 3.3-1 ACTION 2b. with two Reactor Protective System Logic channels inoperable, states, "Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour." ITS 3.3.2 ACTION D requires the affected RTCBs to be opened immediately with two channels of RPS Initiation Logic channel affecting the same trip leg inoperable. This changes Unit 2 CTS 3.3.1 by requiring the affected RTCBs to be opened immediately. The purpose of ITS 3.3.2 ACTION D is to allow for a single failure within a trip leg when one or both trip function channels in the same trip leg fails in a non-trip condition. This change is necessary because with both trip channels of a trip function in the same trip leg inoperable, opening of the affected RTCBs ensures the trip capability is maintained. This change is considered a more restrictive change because a more restrictive completion time is required to open the RTCBs than is required in the CTS.

M06 **Unit 2 only**: CTS 3.3.1 Table 3.3-1 ACTION 5 requires that with one channel of Manual Trip, RTCBs, or RPS Initiation Logic inoperable in MODE 3, 4, or 5 that the inoperable channel be restored within 48 hours or the reactor trip breakers must be opened within the next hour (total of 49 hours). ITS 3.3.2 ACTION C for the same condition that the affected RTCBs be opened in 48 hours. This changes the Unit 2 CTS by changing the time to open the RTCBs from 49 hours to 48 hours. This change is considered a more restrictive change because a more restrictive completion time is required to open the RTCBs than is required in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-1 for RPS instrumentation has three columns stating various requirements for each function. These columns are labeled, "TOTAL NO. OF CHANNELS," "CHANNELS TO TRIP," and "MINIMUM CHANNELS OPERABLE." ITS 3.3.2 does not retain these columns but retains the requirements in the MINIMUM CHANNELS OPERABLE column in the LCO. This changes the CTS by moving the information of the "TOTAL NO. OF CHANNELS" and "CHANNELS TO TRIP" columns to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the number of required channels and the appropriate Condition to enter if a required channel becomes inoperable. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS 3.3.1.1 ACTION 4 and Unit 2 CTS 3.3.1 ACTION 4 states, in part. “one channel may be bypassed for up to 1 hour for surveillance testing.” The equivalent ITS 3.3.2 ACTION B requires the affected RTCBs to be opened in 1 hour. ITS 3.3.2 does not include these procedural details because the channel may continue to be bypassed under the ITS ACTION A. This changes the CTS by moving the procedural detail to the ITS Bases.

The removal of these details, related to methods of surveillance test performance, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specifications Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to methods of surveillance test performance is being removed from the Technical Specifications.

LA03 **Unit 2 only:** *(Type 2 – Removing Descriptions of System Operation)* CTS Table 3.3-1 Footnote (f) states, "Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice." ITS 3.3.2 does not include these system details. This changes the CTS by moving the system detail to the ITS Bases.

The removal of these details, that are related to system operation, from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for the RTCBs to be OPERABLE and the relocated material describes aspects of the RTCB design. In addition, this change is acceptable, because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system operation is being removed from the Technical Specifications.

LA04 **Unit 2 only:** (*Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program*) CTS 3.3.1 Table 3.3-1 ACTION 2.a, states in part, "The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN." ITS 3.3.2 does not include these procedural details. This changes the CTS by moving the operational detail to procedures controlled under the PSL Quality Assurance Topical Report (QATR).

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. OPERABILITY requirements following a unit shutdown are governed by the Technical Specifications MODE requirements, Technical Specification rules of usage (specifically LCO 3.0.4) and operating procedures when transitioning from Cold Shutdown (MODE 5). Also, this change is acceptable because the removed information will be adequately controlled by the requirements of 10 CFR 50, Appendix B, the PSL Quality Assurance Topical Report, and associated implementing procedures. Section XVI, "Corrective Action," of 10 CFR 50, Appendix B requires measures to be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. This regulatory requirement will ensure that the inoperable RPS Initiation Logic channels are restored to OPERABLE status promptly. Changes to the QATR implementing procedures are controlled under 10 CFR 50.54(a)(3) which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA05 **Unit 2 only:** (*Type 2 – Removing Descriptions of System Operation*) CTS 3.3.1 Table 4.3-1 Note 7, states, "The fuse circuitry in the matrix fault protection | R1

circuitry shall be determined to be OPERABLE by testing with the installed test circuitry." ITS SR 3.3.2.2 requires a CHANNEL FUNCTIONAL TEST to be performed on each RPS Initiation logic channel but does not include these procedural/operational details. This changes the CTS by moving the procedural detail to the ITS Bases. | R1

The removal of these details, related to methods of surveillance test performance, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to perform a CHANNEL FUNCTIONAL TEST. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specifications Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to methods of surveillance test performance is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 **Unit 1 only:** *(Category 4 – Relaxation of Required Action)* CTS 3.3.1.1 Table 3.3-1 Action 4, applicable to the Reactor Protective System Logic and Reactor Trip Breaker functions, states " With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours." ITS 3.3.2 ACTION B, applicable in MODES 1 and 2, requires the affected RTCBs to be opened in 1 hour, which can allow continued plant operation. This changes the CTS by requiring the affected RTCBs to be opened rather than requiring a shutdown to HOT STANDBY (MODE 3). | R1

The purpose of ITS 3.3.2 ACTION B is to remove the need for the affected channel to automatically perform its function by manually performing the associated safety function. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. With the RTCB open, the affected Functions are in one-out-of-two logic, which meets the redundancy requirements without requiring shutdown. In this configuration, a single channel failure will not prevent a reactor trip. The Completion Time of 1 hour is adequate to open the affected RTCBs while the risk of having them closed for up to 1 hour is at an acceptable level. This change is designated as less restrictive because the unit shutdown requirement has been revised to allow for opening of the affected RTCBs. Therefore, less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 (*Category 7- Relaxation of Surveillance Frequency*) CTS Table 4.3-1 Functional Unit 12 (Unit 1 – Reactor Protection System Logic and Unit 2 – Reactor Trip

Breakers), including Note 1, requires the performance of a CHANNEL FUNCTIONAL TEST prior to each reactor startup if not performed in the previous 7 days. In addition, CTS Table 4.3-1 also requires the CHANNEL FUNCTIONAL TEST associated with Functional Unit 12 to be performed in accordance with the Surveillance Frequency Control Program (SFCP). ITS SR 3.3.2.2 continues to require the CHANNEL FUNCTIONAL TEST to be performed at a periodic frequency in accordance with the SFCP. However, the ITS does not require this "during startup if not performed in the previous 7 days" test. This changes the CTS by deleting the requirement to perform these startup CHANNEL FUNCTIONALTEST Surveillances.

The purpose of the CTS Table 4.3-1 Functional Unit 12 CHANNEL FUNCTIONAL TEST is to ensure the reactor trip breakers and reactor protection system logic are functioning properly. This change is acceptable because the normal periodic of the CHANNEL FUNCTIONAL TEST Surveillance Frequency in accordance with the SFCP is adequate to ensure these functions are OPERABLE. This change deletes the requirement to perform these startup Surveillances. ITS SR 3.0.4 (CTS 4.0.4) requires the normal periodic Surveillances to be performed and be current prior to entry into the applicable operational conditions. Once the applicable conditions are entered, the normal, periodic Surveillance Frequency provides adequate assurance of OPERABILITY. Therefore, the removal of this Frequency is considered acceptable. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~3~~ 2 Reactor Protective System (RPS) Logic and Trip Initiation ~~(Analog)~~ 2

LCO 3.3.1.1 Table 3.3-1 Functions 1, 12 and 13 DOC M01 DOC A02

LCO 3.3.~~3~~ 2 Six channels of RPS Matrix Logic, four channels of RPS Initiation Logic, ~~[~~four~~]~~ channels of reactor trip circuit breakers (RTCBs), and ~~[~~four~~]~~ channels of Manual Trip shall be OPERABLE. 2 3

Table 3.3-1 Footnote *, Function 1 DOC A03

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5, with any RTCBs closed and any control element assemblies capable of being withdrawn.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01 A. One Matrix Logic channel inoperable. Action 1 OR One Manual Trip channel inoperable in MODE 1 or 2 OR Three Matrix Logic channels inoperable due to a common power source failure de-energizing three matrix power supplies. DOC M01	A.1 Restore channel(s) to OPERABLE status.	48 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~ 1 3
Action 4 DOC L01 B. One channel of ~~Manual Trip,~~ RTCBs, or Initiation Logic inoperable in MODE 1 or 2.	B.1 Open the affected RTCBs.	1 hour 1
Table 3.3-1 Functions 1, 12 and 13 DOC M03 C. One channel of Manual Trip, RTCBs, or Initiation Logic inoperable in MODE 3, 4, or 5.	C.1 Open the affected RTCBs.	48 hours

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M03 D. Two channels of Manual Trip, RTCBs, or Initiation Logic affecting the same trip leg inoperable.	D.1 Open the affected RTCBs.	Immediately
Action 1 DOC M04 E. Required Action and associated Completion Time of Condition A, B, or D not met. OR DOC M04 One or more Functions with two or more Manual Trip, Matrix Logic, Initiation Logic, or RTCB channels inoperable for reasons other than Condition A or D.	E.1 Be in MODE 3. AND E.2 Open all RTCBs.	6 hours 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.3.1.1.1 Table 4.3-1 Function 13 SR 3.3.~~3~~.1	Perform a CHANNEL FUNCTIONAL TEST on each RTCB channel.	~~[[31] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.3.1.1.1 Table 4.3-1 Function 12 DOC L02	SR 3.3.~~3~~.2	Perform a CHANNEL FUNCTIONAL TEST on each RPS Logic channel.	~~[~~ ~~[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
4.3.1.1.1 Table 4.3-1 Function 1	SR 3.3.~~3~~.3	Perform a CHANNEL FUNCTIONAL TEST on each RPS Manual Trip channel.	Once within 7 days prior to each reactor startup
DOC M02	SR 3.3.~~3~~.4	~~[~~Perform a CHANNEL FUNCTIONAL TEST, including separate verification of the undervoltage and shunt trips, on each RTCB channel.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~

CTS

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~3~~ Reactor Protective System (RPS) Logic and Trip Initiation ~~(Analog)~~

LCO 3.3.1 Table 3.3-1 Functions 1, 11 and 12 DOC M01

LCO 3.3.~~3~~ Six channels of RPS Matrix Logic, four channels of RPS Initiation Logic, ~~[~~four~~]~~ channels of reactor trip circuit breakers (RTCBs), and ~~[~~four~~]~~ channels of Manual Trip shall be OPERABLE.

Applicability Table 3.3-1 Footnote *

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5, with any RTCBs closed and any control element assemblies capable of being withdrawn.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01 A. One Matrix Logic channel inoperable. Action 1 OR OR One Manual Trip channel inoperable in MODE 1 or 2 DOC M01 Three Matrix Logic channels inoperable due to a common power source failure de-energizing three matrix power supplies.	A.1 Restore channel(s) to OPERABLE status.	48 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
Table 3.3-1 Action 2 Action 4 DOC M06 B. One channel of ~~Manual Trip,~~ RTCBs, or Initiation Logic inoperable in MODE 1 or 2.	B.1 Open the affected RTCBs.	1 hour
Table 3.3-1 Functions 1, 11, and 12 Action 5 DOC M06 C. One channel of Manual Trip, RTCBs, or Initiation Logic inoperable in MODE 3, 4, or 5.	C.1 Open the affected RTCBs.	48 hours

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action 2.b. (Initiation Logic in Modes 1 & 2) DOC M03 DOC M05 D. Two channels of Manual Trip, RTCBs, or Initiation Logic affecting the same trip leg inoperable.	D.1 Open the affected RTCBs.	Immediately
Actions 1 and 4 DOC M04 E. Required Action and associated Completion Time of Condition A, B, or D not met. OR DOC M04 One or more Functions with two or more Manual Trip, Matrix Logic, Initiation Logic, or RTCB channels inoperable for reasons other than Condition A or D.	E.1 Be in MODE 3. AND E.2 Open all RTCBs.	6 hours 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.3.1.1 Table 4.3-1 Function 12 SR 3.3.~~3~~.1 (2)	Perform a CHANNEL FUNCTIONAL TEST on each RTCB channel.	~~[[31] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2, 3)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.3.1.1 Table 4.3-1 Function 11	SR 3.3.~~3~~.2	Perform a CHANNEL FUNCTIONAL TEST on each RPS Logic channel.	~~[~~ ~~[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
4.3.1.1 Table 4.3-1 Function 1	SR 3.3.~~3~~.3	Perform a CHANNEL FUNCTIONAL TEST on each RPS Manual Trip channel.	Once within 7 days prior to each reactor startup
Table 4.3-1 Function 12 Notation 6	SR 3.3.~~3~~.4	~~[~~ Perform a CHANNEL FUNCTIONAL TEST, including separate verification of the undervoltage and shunt trips, on each RTCB channel.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.2, REACTOR PROTECTION SYSTEM (RPS) LOGIC AND TRIP INITIATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The heading for ISTS 3.3.3 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2). Therefore, ISTS 3.3.2 is not included in the PSL ITS. The Reactor Protective System (RPS) Logic and Trip Initiation (ISTS 3.3.3) is renumbered as ITS 3.3.2.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~3~~ 2 Reactor Protective System (RPS) Logic and Trip Initiation ~~(Analog)~~ 2

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and reactor coolant pressure boundary integrity during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling,
- Fuel centerline melting shall not occur, and
- The Reactor Coolant System pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the ~~10 CFR 50 (Ref. 1) and~~ 10 CFR ~~100~~ (Ref. ~~2~~) criteria during AOOs. 50.67 1 1

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~ (Ref. ~~2~~) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event. 1 50.67 1

BASES

BACKGROUND (continued)

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels,
- Bistable trip units,
- RPS Logic, and
- Reactor trip circuit breakers (RTCBs).

This LCO addresses the RPS Logic and RTCBs, including Manual Trip capability. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating," provides a description of the role of this equipment in the RPS. This is summarized below:

RPS Logic

The RPS Logic, consisting of Matrix and Initiation Logic, employs a scheme that provides a reactor trip when bistables in any two of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic. ~~This logic and the RTCB configuration are shown in Figure B 3.3.1-1.~~ 1 5

(one out of two taken twice)

Bistable relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized RTCB control relays (K1, K2, K3, and K4). The trip paths thus each have six contacts in series, one from each matrix, and perform a logical OR function, opening the RTCBs if any one or more of the six logic matrices indicate a coincidence condition.

BASES

BACKGROUND (continued)

Each trip path is responsible for opening one set of two of the eight RTCBs. The RTCB control relays (K-relays), when de-energized, interrupt power to the breaker undervoltage trip attachments and simultaneously apply power to the shunt trip attachments on each of the two breakers. Actuation of either the undervoltage or shunt trip attachment is sufficient to open the RTCB and interrupt power from the motor generator (MG) sets to the control element drive mechanisms (CEDMs).

When a coincidence occurs in two RPS channels, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four breaker control relays, which simultaneously de-energize the undervoltage and energize the shunt trip attachments in all eight RTCBs, tripping them open.

The Initiation Logic consists of the trip path power source, matrix relays and their associated contacts, all interconnecting wiring, and solid state (auxiliary) relays through the K-relay contacts in the RTCB control circuitry.

It is possible to change the two-out-of-four RPS Logic to a two-out-of-three logic for a given input parameter in one channel at a time by trip channel bypassing select portions of the matrix logic. Trip channel bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional channels indicate a trip condition. Trip channel bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. An interlock prevents simultaneous trip channel bypassing of the same parameter in more than one channel. Trip channel bypassing is normally employed during maintenance or testing.

Reactor Trip Circuit Breakers (RTCBs)

The reactor trip switchgear~~, shown in Figure B 3.3.1-1,~~ consists of eight RTCBs, which are operated in four sets of two breakers (four channels). Power input to the reactor trip switchgear comes from two full capacity MG sets operated in parallel such that the loss of either MG set does not de-energize the CEDMs. There are two separate CEDM power supply 5

BASES

BACKGROUND (continued)

buses, each bus powering half of the CEDMs. Power is supplied from the MG sets to each bus via two redundant paths (trip legs). Trip legs 1A and 1B supply power to CEDM bus 1. Trip legs 2A and 2B supply power to CEDM bus 2. This ensures that a fault or the opening of a breaker in one trip leg (i.e., for testing purposes) will not interrupt power to the CEDM buses.

Each of the four trip legs consists of two RTCBs in series. The two RTCBs within a trip leg are actuated by separate initiation circuits.

The eight RTCBs are operated as four sets of two breakers (four channels). For example, if a breaker receives an open signal in trip leg A (for CEDM bus 1), an identical breaker in trip leg B (for CEDM bus 2) will also receive an open signal. This arrangement ensures that power is interrupted to both CEDM buses, thus preventing trip of only half of the control element assemblies (CEAs) (a half trip). Any one inoperable breaker in a channel will make the entire channel inoperable.

Each set of RTCBs is operated by either a Manual Trip push button or an RPS actuated K-relay. There are four Manual Trip push buttons, arranged in two sets of two~~, as shown in Figure B 3.3.1-1~~. Depressing both push buttons in either set will result in a reactor trip. 5

When a Manual Trip is initiated using the control room push buttons, the RPS trip paths and K-relays are bypassed, and the RTCB undervoltage and shunt trip attachments are actuated independent of the RPS.

Manual Trip circuitry includes the push button and interconnecting wiring to both RTCBs necessary to actuate both the undervoltage and shunt trip attachments, but excludes the K-relay contacts and their interconnecting wiring to the RTCBs, which are considered part of the Initiation Logic.

Functional testing of the entire RPS, from bistable input through the opening of individual sets of RTCBs, can be performed either at power or shutdown ~~and is normally performed on a quarterly basis~~. FSAR, Section ~~[~~7.2~~]~~ (Ref. ~~3~~), explains RPS testing in more detail. 1 3

in accordance with the Surveillance Frequency Control Program 2

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APPLICABLE SAFETY ANALYSES

Reactor Protective System (RPS) Logic

The RPS Logic provides for automatic trip initiation to maintain the SLs during AOOs and assist the ESF systems in ensuring acceptable consequences during accidents. All transients and accidents that call for a reactor trip assume the RPS Logic is functioning as designed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reactor Trip Circuit Breakers (RTCBs)

All of the transient and accident analyses that call for a reactor trip assume that the RTCBs operate and interrupt power to the CEDMs.

Manual Trip

There are no accident analyses that take credit for the Manual Trip; however, the Manual Trip is part of the RPS circuitry. It is used by the operator to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint. A Manual Trip accomplishes the same results as any one of the automatic trip Functions.

The RPS Logic and initiation satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Reactor Protective System (RPS) Logic

Failures of individual bistable relays and their contacts are addressed in LCO 3.3.1. This Specification addresses failures of the Matrix Logic not addressed in the above, such as the failure of matrix relay power supplies or the failure of the trip channel bypass contact in the bypass condition.

instrument

Loss of a single ~~vital~~ bus will de-energize one of the two power supplies in each of three matrices. This will result in four RTCBs opening; however, the remaining four closed RTCBs will prevent a reactor trip. For the purposes of this LCO, de-energizing up to three matrix power supplies due to a single failure is to be treated as a single channel failure, providing the affected matrix relays de-energize as designed, opening the affected RTCBs. 1

Each of the four Initiation Logic channels opens one set of RTCBs if any of the six coincidence matrices de-energize their associated matrix relays. They thus perform a logical OR function. Each Initiation Logic channel has its own power supply and is independent of the others. An Initiation Logic channel includes the matrix relay through to the K-relay contacts, which open the RTCB.

It is possible for two Initiation Logic channels affecting the same trip leg to de-energize if a matrix power supply or ~~vital~~ instrument bus fails. This will result in opening the two affected sets of RTCBs. 1

BASES

LCO (continued)

If one set of RTCBs has been opened in response to a single RTCB channel, Initiation Logic channel, or Manual Trip channel failure, the affected set of RTCBs may be closed for up to 1 hour for Surveillance on the OPERABLE Initiation Logic, RTCB, and Manual Trip channels. In this case, the redundant set of RTCBs will provide protection if a trip should be required. It is unlikely that a trip will be required during the Surveillance, coincident with a failure of the remaining series RTCB channel. If a single matrix power supply or ~~vital~~ bus failure has opened two sets of RTCBs, Manual Trip and RTCB testing on the closed breakers cannot be performed without causing a trip.

instrument

1

1. Matrix Logic

   This LCO requires six channels of Matrix Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any RTCB is closed and any CEA is capable of being withdrawn.

2. Initiation Logic

   This LCO requires four channels of Initiation Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any RTCB is closed and any CEA is capable of being withdrawn.

3. Reactor Trip Circuit Breakers (RTCBs)

   The LCO requires four RTCB channels to be OPERABLE in MODES 1 and 2, as well as in MODES 3, 4, and 5 when any RTCB is closed and any CEA is capable of being withdrawn.

   Each channel consists of two breakers operated in a single set by the Initiation Logic or Manual Trip circuitry. This ensures that power is interrupted at identical locations in the trip legs for both CEDM buses, thus preventing power removal to only one CEDM bus (a half trip).

   Failure of a single breaker affects the entire channel, and both breakers in the set must be opened. Without reliable RTCBs and associated support circuitry, a reactor trip cannot occur whether initiated automatically or manually.

   Each channel of RTCBs starts at the contacts actuated by the K-relay, and the contacts actuated by the Manual Trip, for each set of breakers. The K-relay actuated contacts and the upstream circuitry are considered to be RPS Logic. Manual Trip contacts and upstream circuitry are considered to be Manual Trip circuitry.

BASES

LCO (continued)

I

~~A Note associated with the ACTIONS states that~~ If one set of RTCBs has been opened in response to a single RTCB channel, Initiation Logic channel, or Manual Trip channel failure, the affected set of RTCBs may be closed for up to 1 hour for Surveillance on the OPERABLE Initiation Logic, RTCB, and Manual Trip channels. In this case, the redundant set of RTCBs will provide protection. If a single matrix power supply or vital bus failure has opened two sets of RTCBs, Manual Trip and RTCB testing on the closed breakers cannot be performed without causing a trip. ~~This Note is not applicable to Condition A, with one Matrix Logic channel inoperable.~~ 1 1

4. Manual Trip

The LCO requires all four Manual Trip channels to be OPERABLE in MODES 1 and 2, and MODES 3, 4, and 5 when any RTCB is closed and any CEA is capable of being withdrawn.

Two independent sets of two adjacent push buttons are provided at separate locations. Each push button is considered a channel and operates two of the eight RTCBs. Depressing both push buttons in either set will cause an interruption of power to the CEDMs, allowing the CEAs to fall into the core. This design ensures that no single failure in any push button circuit can either cause or prevent a reactor trip.

APPLICABILITY

The RPS Matrix Logic, RTCBs, and Manual Trip are required to be OPERABLE in any MODE when any CEA is capable of being withdrawn from the core (i.e., RTCBs closed and power available to the CEDMs). This ensures the reactor can be tripped when necessary, but allows for maintenance and testing when the reactor trip is not needed.

In MODES 3, 4, and 5 with all the RTCBs open, the CEAs are not capable of withdrawal and these Functions do not have to be OPERABLE. However, two ~~[~~logarithmic~~]~~ ~~power~~ level channels must be OPERABLE to ensure proper indication of neutron population and to indicate a boron dilution event. This is addressed in LCO 3.3.~~13~~, "~~[~~Logarithmic~~]~~ ~~Power~~ Monitoring Channels."

neutron flux 1 3 1

Neutron Flux 11

ACTIONS

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

BASES

ACTIONS (continued)

A.1

, one Manual Trip channel is inoperable, 1

Condition A applies if one Matrix Logic channel is inoperable or three Matrix Logic channels are inoperable due to a common power source failure de-energizing three matrix power supplies, in any applicable MODE. Loss of a single ~~vital~~ instrument bus will de-energize one of the two matrix power supplies in up to three matrices. This is considered a single matrix failure, providing the matrix relays associated with the failed power supplies de-energize as required. 1

Failure of the matrix relays to de-energize in all three affected matrices could, when combined with trip channel bypassing of bistable relay contacts in the other matrices, result in loss of RPS function.

The channel must be restored to OPERABLE status within 48 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The Completion Time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second Matrix Logic channel is low during any given 48 hour interval. If the channel cannot be restored to OPERABLE status within 48 hours, Condition E is entered. 3

B.1

or

Condition B applies to one Initiation Logic channel, RTCB channel~~, or Manual Trip channel~~ in MODES 1 and 2, since they have the same actions. MODES 3, 4, and 5, with the RTCBs shut, are addressed in Condition C. These Required Actions require opening the affected RTCBs. This removes the need for the affected channel by performing its associated safety function. With the RTCB open, the affected Functions are in one-out-of-two logic, which meets redundancy requirements, but testing on the OPERABLE channels cannot be performed without causing a reactor trip unless the RTCBs in the inoperable channels are closed to permit testing. 1

The inoperable channel may be bypassed during the one hour period to perform testing, if necessary, provided the RTCBs are opened within the required Completion Time.

Required Action B.1 provides for opening the RTCBs associated with the inoperable channel within a Completion Time of 1 hour. This Required Action is conservative, since depressing the Manual Trip push button associated with either set of breakers in the other trip leg will cause a reactor trip. With this configuration, a single channel failure will not prevent a reactor trip. The allotted Completion Time is adequate to open the affected RTCBs while maintaining the risk of having them closed at an acceptable level. 1

BASES

ACTIONS (continued)

C.1

Condition C applies to the failure of one Initiation Logic channel, RTCB channel, or Manual Trip channel ~~affecting the same trip leg~~ in MODE 3, 4, or 5 with the RTCBs closed. The channel must be restored to OPERABLE status within 48 hours. If the inoperable channel cannot be restored to OPERABLE status within 48 hours, the affected RTCBs must be opened. ~~In some cases, this condition may effect all of the RTCBs.~~ This removes the need for the affected channel by performing its associated safety function. With the RTCBs open, the affected functions are in a one-out-of-two logic, which meets redundancy requirements. 1 1

The Completion Time of 48 hours is consistent with that of other RPS instrumentation and should be adequate to repair most failures.

Testing on the OPERABLE channels cannot be performed without causing a reactor trip unless the RTCBs in the inoperable channels are closed to permit testing.

D.1

Condition D applies to the failure of both Manual Trip or Initiation Logic channels affecting the same trip leg. Since this will open two channels of RTCBs, this Condition is also applicable to the two affected channels of RTCBs. This Condition allows for loss of a single ~~vital~~ instrument bus or matrix power supply, which will de-energize both Initiation Logic channels in the same trip leg. This will open both sets of RTCBs in the affected trip leg, satisfying the Required Action of opening the affected RTCBs. 1

Of greater concern is the failure of the initiation circuit in a nontrip condition (e.g., due to two initiation K-relay failures). With only one Initiation Logic channel failed in a nontrip condition, there is still the redundant set of RTCBs in the trip leg. With both failed in a nontrip condition, the reactor will not trip automatically when required. In either case, the affected RTCBs must be opened immediately by using the appropriate Manual Trip push buttons, since each of the four push buttons opens one set of RTCBs, independent of the initiation circuitry. Caution must be exercised, since depressing the wrong push buttons may result in a reactor trip.

BASES

ACTIONS (continued)

If two Manual Trip channels are inoperable and affecting the same trip leg, the associated RTCBs must be opened immediately to ensure Manual Trip capability is maintained. With the affected RTCBs open, any one of two Manual Trip push buttons being depressed will result in a reactor trip.

If the affected RTCB(s) cannot be opened, Condition E is entered. This would only occur if there is a failure in the Manual Trip circuitry or the RTCB(s).

E.1 and E.2

Condition E is entered if Required Actions associated with Condition A, B, or D are not met within the required Completion Time or if for one or more Functions more than one Manual Trip, Matrix Logic, Initiation Logic, or RTCB channel is inoperable for reasons other than Condition A or D.

If the RTCBs associated with the inoperable channel cannot be opened, the reactor must be shut down within 6 hours and all the RTCBs opened. A Completion Time of 6 hours is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems and to open RTCBs. All RTCBs should then be opened, placing the plant in a MODE where the LCO does not apply and ensuring no CEA withdrawal occurs.

SURVEILLANCE REQUIREMENTS

~~----------REVIEWER'S NOTE----------~~
~~In order for a plant to take credit for topical reports as the basis for justifying Frequencies, topical reports must be supported by an NRC staff Safety Evaluation Report that establishes the acceptability of each topical report for that unit (Ref. 4).~~
~~----------~~ 4

2
SR 3.3.~~3~~.1 2

A CHANNEL FUNCTIONAL TEST is performed on each RTCB channel. This verifies proper operation of each RTCB. The RTCB must then be closed prior to testing the other RTCBs, or a reactor trip may result~~. [The Frequency of 31 days is based on the reliability analysis presented in Topical Report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation," (Ref. 5).~~ 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4

2

SR 3.3.~~3~~.2 2

A CHANNEL FUNCTIONAL TEST on each RPS Logic channel is performed to ensure the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

In addition to power supply tests, the RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference ~~3~~. 2 1
These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. The first test, the bistable test, is addressed by SR 3.3.1.4 in LCO 3.3.1.

This SR addresses the two tests associated with the RPS Logic: Matrix Logic and Trip Path.

Matrix Logic Tests

These tests are performed one matrix at a time. They verify that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. The Matrix Logic tests will detect any short circuits around the bistable contacts in the coincidence logic such as may be caused by faulty bistable relay or trip channel bypass contacts.

SURVEILLANCE REQUIREMENTS (continued)

Trip Path Tests

These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, opening the affected set of RTCBs. The RTCBs must then be closed prior to testing the other three initiation circuits, or a reactor trip may result.

~~[The Frequency of [92] days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 5).~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4

~~]~~ 3

2

SR 3.3.~~3~~.3 2

A CHANNEL FUNCTIONAL TEST on the Manual Trip channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Manual Trip Function can be tested either at power or shutdown. However, the simplicity of this circuitry and the absence of drift concern makes this Frequency adequate. Additionally, operating experience has shown that these components usually pass the Surveillance when performed once within 7 days prior to each reactor startup.

BASES

SURVEILLANCE REQUIREMENTS (continued)

2

~~[~~ SR 3.3.~~3~~.4 2 3

Each RTCB is actuated by an undervoltage coil and a shunt trip coil. The system is designed so that either de-energizing the undervoltage coil or energizing the shunt trip coil will cause the circuit breaker to open. When an RTCB is opened, either during an automatic reactor trip or by using the manual push buttons in the control room, the undervoltage coil is de-energized and the shunt trip coil is energized. This makes it impossible to determine if one of the coils or associated circuitry is defective.

Therefore, a CHANNEL FUNCTIONAL TEST is performed that individually tests all four sets of undervoltage coils and all four sets of shunt trip coils. During undervoltage coil testing, the shunt trip coils shall remain de-energized, preventing their operation. Conversely, during shunt trip coil testing, the undervoltage coils shall remain energized, preventing their operation. This Surveillance ensures that every undervoltage coil and every shunt trip coil is capable of performing its intended function and that no single active failure of any RTCB component will prevent a reactor trip. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. ~~[The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the Frequency of once every 18 months.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------~~ ~~]~~ 4 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

If one set of RTCBs has been opened in response to a single RTCB channel, Initiation Logic channel, or Manual Trip channel failure, the affected set of RTCBs may be closed for up to 1 hour for Surveillance on the OPERABLE Initiation Logic, RTCB, and Manual Trip channels. In this case, the redundant set of RTCBs will provide protection if a trip should be required. It is unlikely that a trip will be required during the Surveillance, coincident with a failure of the remaining series RTCB channel. If a single matrix power supply or vital bus failure has opened two sets of RTCBs, Manual Trip and RTCB testing on the closed breakers cannot be performed without causing a trip.~~]~~ 3

REFERENCES

~~1. 10 CFR 50, Appendix A.~~

1 2. 10 CFR ~~10~~0. 50.67

2 ~~3~~. FSAR, Section ~~[~~7.2~~]~~.
U

~~4. NRC Safety Evaluation Report, [Date].~~

~~5. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989.~~

1 3

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~3~~ 2 Reactor Protective System (RPS) Logic and Trip Initiation ~~(Analog)~~ 2

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and reactor coolant pressure boundary integrity during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling,
- Fuel centerline melting shall not occur, and
- The Reactor Coolant System pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the ~~10 CFR 50 (Ref. 1) and~~ 10 CFR ~~100~~ (Ref. ~~2~~) criteria during AOOs. 50.67 1 1

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~ (Ref. ~~2~~) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event. 1 50.67 1

BASES

BACKGROUND (continued)

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels,
- Bistable trip units,
- RPS Logic, and
- Reactor trip circuit breakers (RTCBs).

This LCO addresses the RPS Logic and RTCBs, including Manual Trip capability. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating," provides a description of the role of this equipment in the RPS. This is summarized below:

RPS Logic

The RPS Logic, consisting of Matrix and Initiation Logic, employs a scheme that provides a reactor trip when bistables in any two of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic. ~~This logic and the RTCB configuration are shown in Figure B 3.3.1-1.~~ 1 5

(one out of two taken twice)

Bistable relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized RTCB control relays (K1, K2, K3, and K4). The trip paths thus each have six contacts in series, one from each matrix, and perform a logical OR function, opening the RTCBs if any one or more of the six logic matrices indicate a coincidence condition.

BASES

BACKGROUND (continued)

Each trip path is responsible for opening one set of two of the eight RTCBs. The RTCB control relays (K-relays), when de-energized, interrupt power to the breaker undervoltage trip attachments and simultaneously apply power to the shunt trip attachments on each of the two breakers. Actuation of either the undervoltage or shunt trip attachment is sufficient to open the RTCB and interrupt power from the motor generator (MG) sets to the control element drive mechanisms (CEDMs).

When a coincidence occurs in two RPS channels, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four breaker control relays, which simultaneously de-energize the undervoltage and energize the shunt trip attachments in all eight RTCBs, tripping them open.

The Initiation Logic consists of the trip path power source, matrix relays and their associated contacts, all interconnecting wiring, and solid state (auxiliary) relays through the K-relay contacts in the RTCB control circuitry.

It is possible to change the two-out-of-four RPS Logic to a two-out-of-three logic for a given input parameter in one channel at a time by trip channel bypassing select portions of the matrix logic. Trip channel bypassing a bistable effectively shorts the bistable relay contacts in the three matrices associated with that channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional channels indicate a trip condition. Trip channel bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. An interlock prevents simultaneous trip channel bypassing of the same parameter in more than one channel. Trip channel bypassing is normally employed during maintenance or testing.

Reactor Trip Circuit Breakers (RTCBs)

The reactor trip switchgear~~, shown in Figure B 3.3.1-1,~~ consists of eight RTCBs, which are operated in four sets of two breakers (four channels). Power input to the reactor trip switchgear comes from two full capacity MG sets operated in parallel such that the loss of either MG set does not de-energize the CEDMs. There are two separate CEDM power supply 5

BASES

BACKGROUND (continued)

buses, each bus powering half of the CEDMs. Power is supplied from the MG sets to each bus via two redundant paths (trip legs). Trip legs 1A and 1B supply power to CEDM bus 1. Trip legs 2A and 2B supply power to CEDM bus 2. This ensures that a fault or the opening of a breaker in one trip leg (i.e., for testing purposes) will not interrupt power to the CEDM buses.

Each of the four trip legs consists of two RTCBs in series. The two RTCBs within a trip leg are actuated by separate initiation circuits.

The eight RTCBs are operated as four sets of two breakers (four channels). For example, if a breaker receives an open signal in trip leg A (for CEDM bus 1), an identical breaker in trip leg B (for CEDM bus 2) will also receive an open signal. This arrangement ensures that power is interrupted to both CEDM buses, thus preventing trip of only half of the control element assemblies (CEAs) (a half trip). Any one inoperable breaker in a channel will make the entire channel inoperable.

Each set of RTCBs is operated by either a Manual Trip push button or an RPS actuated K-relay. There are four Manual Trip push buttons, arranged in two sets of two~~, as shown in Figure B 3.3.1-1~~. Depressing both push buttons in either set will result in a reactor trip. 5

When a Manual Trip is initiated using the control room push buttons, the RPS trip paths and K-relays are bypassed, and the RTCB undervoltage and shunt trip attachments are actuated independent of the RPS.

Manual Trip circuitry includes the push button and interconnecting wiring to both RTCBs necessary to actuate both the undervoltage and shunt trip attachments, but excludes the K-relay contacts and their interconnecting wiring to the RTCBs, which are considered part of the Initiation Logic.

Functional testing of the entire RPS, from bistable input through the opening of individual sets of RTCBs, can be performed either at power or shutdown ~~and is normally performed on a quarterly basis~~. FSAR, Section ~~[~~7.2~~]~~ (Ref. ~~3~~), explains RPS testing in more detail. 1 3

in accordance with the Surveillance Frequency Control Program

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APPLICABLE SAFETY ANALYSES

Reactor Protective System (RPS) Logic

The RPS Logic provides for automatic trip initiation to maintain the SLs during AOOs and assist the ESF systems in ensuring acceptable consequences during accidents. All transients and accidents that call for a reactor trip assume the RPS Logic is functioning as designed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reactor Trip Circuit Breakers (RTCBs)

All of the transient and accident analyses that call for a reactor trip assume that the RTCBs operate and interrupt power to the CEDMs.

Manual Trip

There are no accident analyses that take credit for the Manual Trip; however, the Manual Trip is part of the RPS circuitry. It is used by the operator to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint. A Manual Trip accomplishes the same results as any one of the automatic trip Functions.

The RPS Logic and initiation satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Reactor Protective System (RPS) Logic

Failures of individual bistable relays and their contacts are addressed in LCO 3.3.1. This Specification addresses failures of the Matrix Logic not addressed in the above, such as the failure of matrix relay power supplies or the failure of the trip channel bypass contact in the bypass condition.

instrument

Loss of a single ~~vital~~ bus will de-energize one of the two power supplies in each of three matrices. This will result in four RTCBs opening; however, the remaining four closed RTCBs will prevent a reactor trip. For the purposes of this LCO, de-energizing up to three matrix power supplies due to a single failure is to be treated as a single channel failure, providing the affected matrix relays de-energize as designed, opening the affected RTCBs. 1

Each of the four Initiation Logic channels opens one set of RTCBs if any of the six coincidence matrices de-energize their associated matrix relays. They thus perform a logical OR function. Each Initiation Logic channel has its own power supply and is independent of the others. An Initiation Logic channel includes the matrix relay through to the K-relay contacts, which open the RTCB.

It is possible for two Initiation Logic channels affecting the same trip leg to de-energize if a matrix power supply or ~~vital~~ instrument bus fails. This will result in opening the two affected sets of RTCBs. 1

BASES

LCO (continued)

If one set of RTCBs has been opened in response to a single RTCB channel, Initiation Logic channel, or Manual Trip channel failure, the affected set of RTCBs may be closed for up to 1 hour for Surveillance on the OPERABLE Initiation Logic, RTCB, and Manual Trip channels. In this case, the redundant set of RTCBs will provide protection if a trip should be required. It is unlikely that a trip will be required during the Surveillance, coincident with a failure of the remaining series RTCB channel. If a single matrix power supply or ~~vital~~ bus failure has opened two sets of RTCBs, Manual Trip and RTCB testing on the closed breakers cannot be performed without causing a trip.

instrument

1

1. Matrix Logic

   This LCO requires six channels of Matrix Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any RTCB is closed and any CEA is capable of being withdrawn.

2. Initiation Logic

   This LCO requires four channels of Initiation Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any RTCB is closed and any CEA is capable of being withdrawn.

3. Reactor Trip Circuit Breakers (RTCBs)

   The LCO requires four RTCB channels to be OPERABLE in MODES 1 and 2, as well as in MODES 3, 4, and 5 when any RTCB is closed and any CEA is capable of being withdrawn.

   Each channel consists of two breakers operated in a single set by the Initiation Logic or Manual Trip circuitry. This ensures that power is interrupted at identical locations in the trip legs for both CEDM buses, thus preventing power removal to only one CEDM bus (a half trip).

   Failure of a single breaker affects the entire channel, and both breakers in the set must be opened. Without reliable RTCBs and associated support circuitry, a reactor trip cannot occur whether initiated automatically or manually.

   Each channel of RTCBs starts at the contacts actuated by the K-relay, and the contacts actuated by the Manual Trip, for each set of breakers. The K-relay actuated contacts and the upstream circuitry are considered to be RPS Logic. Manual Trip contacts and upstream circuitry are considered to be Manual Trip circuitry.

BASES

LCO (continued)

I

~~A Note associated with the ACTIONS states that~~ If one set of RTCBs has been opened in response to a single RTCB channel, Initiation Logic channel, or Manual Trip channel failure, the affected set of RTCBs may be closed for up to 1 hour for Surveillance on the OPERABLE Initiation Logic, RTCB, and Manual Trip channels. In this case, the redundant set of RTCBs will provide protection. If a single matrix power supply or vital bus failure has opened two sets of RTCBs, Manual Trip and RTCB testing on the closed breakers cannot be performed without causing a trip. ~~This Note is not applicable to Condition A, with one Matrix Logic channel inoperable.~~ 1 1

4. Manual Trip

The LCO requires all four Manual Trip channels to be OPERABLE in MODES 1 and 2, and MODES 3, 4, and 5 when any RTCB is closed and any CEA is capable of being withdrawn.

Two independent sets of two adjacent push buttons are provided at separate locations. Each push button is considered a channel and operates two of the eight RTCBs. Depressing both push buttons in either set will cause an interruption of power to the CEDMs, allowing the CEAs to fall into the core. This design ensures that no single failure in any push button circuit can either cause or prevent a reactor trip.

APPLICABILITY

The RPS Matrix Logic, RTCBs, and Manual Trip are required to be OPERABLE in any MODE when any CEA is capable of being withdrawn from the core (i.e., RTCBs closed and power available to the CEDMs). This ensures the reactor can be tripped when necessary, but allows for maintenance and testing when the reactor trip is not needed.

In MODES 3, 4, and 5 with all the RTCBs open, the CEAs are not capable of withdrawal and these Functions do not have to be OPERABLE. However, two ~~[~~logarithmic~~]~~ ~~power~~ level channels must be OPERABLE to ensure proper indication of neutron population and to indicate a boron dilution event. This is addressed in LCO 3.3.~~13~~, "~~[~~Logarithmic~~]~~ ~~Power~~ Monitoring Channels."

neutron flux 1 3 1

Neutron Flux 11

ACTIONS

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

BASES

ACTIONS (continued)

A.1

, one Manual Trip channel is inoperable, 1

Condition A applies if one Matrix Logic channel is inoperable or three Matrix Logic channels are inoperable due to a common power source failure de-energizing three matrix power supplies, in any applicable MODE. Loss of a single ~~vital~~ instrument bus will de-energize one of the two matrix power supplies in up to three matrices. This is considered a single matrix failure, providing the matrix relays associated with the failed power supplies de-energize as required. 1

Failure of the matrix relays to de-energize in all three affected matrices could, when combined with trip channel bypassing of bistable relay contacts in the other matrices, result in loss of RPS function.

The channel must be restored to OPERABLE status within 48 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The Completion Time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second Matrix Logic channel is low during any given 48 hour interval. If the channel cannot be restored to OPERABLE status within 48 hours, Condition E is entered. 3

B.1

or

Condition B applies to one Initiation Logic channel, RTCB channel~~, or Manual Trip channel~~ in MODES 1 and 2, since they have the same actions. MODES 3, 4, and 5, with the RTCBs shut, are addressed in Condition C. These Required Actions require opening the affected RTCBs. This removes the need for the affected channel by performing its associated safety function. With the RTCB open, the affected Functions are in one-out-of-two logic, which meets redundancy requirements, but testing on the OPERABLE channels cannot be performed without causing a reactor trip unless the RTCBs in the inoperable channels are closed to permit testing. 1

The inoperable channel may be bypassed during the one hour period to perform testing, if necessary, provided the RTCBs are opened within the required Completion Time.

Required Action B.1 provides for opening the RTCBs associated with the inoperable channel within a Completion Time of 1 hour. This Required Action is conservative, since depressing the Manual Trip push button associated with either set of breakers in the other trip leg will cause a reactor trip. With this configuration, a single channel failure will not prevent a reactor trip. The allotted Completion Time is adequate to open the affected RTCBs while maintaining the risk of having them closed at an acceptable level.

BASES

ACTIONS (continued)

C.1

Condition C applies to the failure of one Initiation Logic channel, RTCB channel, or Manual Trip channel ~~affecting the same trip leg~~ in MODE 3, 4, or 5 with the RTCBs closed. The channel must be restored to OPERABLE status within 48 hours. If the inoperable channel cannot be restored to OPERABLE status within 48 hours, the affected RTCBs must be opened. ~~In some cases, this condition may effect all of the RTCBs.~~ This removes the need for the affected channel by performing its associated safety function. With the RTCBs open, the affected functions are in a one-out-of-two logic, which meets redundancy requirements. 1 1

The Completion Time of 48 hours is consistent with that of other RPS instrumentation and should be adequate to repair most failures.

Testing on the OPERABLE channels cannot be performed without causing a reactor trip unless the RTCBs in the inoperable channels are closed to permit testing.

D.1

Condition D applies to the failure of both Manual Trip or Initiation Logic channels affecting the same trip leg. Since this will open two channels of RTCBs, this Condition is also applicable to the two affected channels of RTCBs. This Condition allows for loss of a single ~~vital~~ instrument bus or matrix power supply, which will de-energize both Initiation Logic channels in the same trip leg. This will open both sets of RTCBs in the affected trip leg, satisfying the Required Action of opening the affected RTCBs. 1

Of greater concern is the failure of the initiation circuit in a nontrip condition (e.g., due to two initiation K-relay failures). With only one Initiation Logic channel failed in a nontrip condition, there is still the redundant set of RTCBs in the trip leg. With both failed in a nontrip condition, the reactor will not trip automatically when required. In either case, the affected RTCBs must be opened immediately by using the appropriate Manual Trip push buttons, since each of the four push buttons opens one set of RTCBs, independent of the initiation circuitry. Caution must be exercised, since depressing the wrong push buttons may result in a reactor trip.

BASES

ACTIONS (continued)

If two Manual Trip channels are inoperable and affecting the same trip leg, the associated RTCBs must be opened immediately to ensure Manual Trip capability is maintained. With the affected RTCBs open, any one of two Manual Trip push buttons being depressed will result in a reactor trip.

If the affected RTCB(s) cannot be opened, Condition E is entered. This would only occur if there is a failure in the Manual Trip circuitry or the RTCB(s).

E.1 and E.2

Condition E is entered if Required Actions associated with Condition A, B, or D are not met within the required Completion Time or if for one or more Functions more than one Manual Trip, Matrix Logic, Initiation Logic, or RTCB channel is inoperable for reasons other than Condition A or D.

If the RTCBs associated with the inoperable channel cannot be opened, the reactor must be shut down within 6 hours and all the RTCBs opened. A Completion Time of 6 hours is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems and to open RTCBs. All RTCBs should then be opened, placing the plant in a MODE where the LCO does not apply and ensuring no CEA withdrawal occurs.

SURVEILLANCE REQUIREMENTS

~~----------REVIEWER'S NOTE----------~~
~~In order for a plant to take credit for topical reports as the basis for justifying Frequencies, topical reports must be supported by an NRC staff Safety Evaluation Report that establishes the acceptability of each topical report for that unit (Ref. 4).~~
~~----------~~ 4

2
SR 3.3.~~3~~.1 2

A CHANNEL FUNCTIONAL TEST is performed on each RTCB channel. This verifies proper operation of each RTCB. The RTCB must then be closed prior to testing the other RTCBs, or a reactor trip may result~~. [The Frequency of 31 days is based on the reliability analysis presented in Topical Report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation," (Ref. 5).~~ 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.3.~~3~~.2

A CHANNEL FUNCTIONAL TEST on each RPS Logic channel is performed to ensure the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

In addition to power supply tests, the RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference ~~3~~. These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. The first test, the bistable test, is addressed by SR 3.3.1.4 in LCO 3.3.1.

This SR addresses the two tests associated with the RPS Logic: Matrix Logic and Trip Path.

Matrix Logic Tests

These tests are performed one matrix at a time. They verify that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. The Matrix Logic tests will detect any short circuits around the bistable contacts in the coincidence logic such as may be caused by faulty bistable relay or trip channel bypass contacts.

The matrix logic includes fuses in all matrix interbay connections as part of the fault protection circuitry. A test circuit is provided for checking the fuses associated with the matrix fault protection circuitry.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Trip Path Tests

These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, opening the affected set of RTCBs. The RTCBs must then be closed prior to testing the other three initiation circuits, or a reactor trip may result.

~~[The Frequency of [92] days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 5).~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.3.~~3~~.3

A CHANNEL FUNCTIONAL TEST on the Manual Trip channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Manual Trip Function can be tested either at power or shutdown. However, the simplicity of this circuitry and the absence of drift concern makes this Frequency adequate. Additionally, operating experience has shown that these components usually pass the Surveillance when performed once within 7 days prior to each reactor startup.

BASES

SURVEILLANCE REQUIREMENTS (continued)

2

~~[~~ SR 3.3.~~3~~.4 3 2

Each RTCB is actuated by an undervoltage coil and a shunt trip coil. The system is designed so that either de-energizing the undervoltage coil or energizing the shunt trip coil will cause the circuit breaker to open. When an RTCB is opened, either during an automatic reactor trip or by using the manual push buttons in the control room, the undervoltage coil is de-energized and the shunt trip coil is energized. This makes it impossible to determine if one of the coils or associated circuitry is defective.

Therefore, a CHANNEL FUNCTIONAL TEST is performed that individually tests all four sets of undervoltage coils and all four sets of shunt trip coils. During undervoltage coil testing, the shunt trip coils shall remain de-energized, preventing their operation. Conversely, during shunt trip coil testing, the undervoltage coils shall remain energized, preventing their operation. This Surveillance ensures that every undervoltage coil and every shunt trip coil is capable of performing its intended function and that no single active failure of any RTCB component will prevent a reactor trip. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. ~~[The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the Frequency of once every 18 months.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------~~ ~~]~~ 4 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

If one set of RTCBs has been opened in response to a single RTCB channel, Initiation Logic channel, or Manual Trip channel failure, the affected set of RTCBs may be closed for up to 1 hour for Surveillance on the OPERABLE Initiation Logic, RTCB, and Manual Trip channels. In this case, the redundant set of RTCBs will provide protection if a trip should be required. It is unlikely that a trip will be required during the Surveillance, coincident with a failure of the remaining series RTCB channel. If a single matrix power supply or vital bus failure has opened two sets of RTCBs, Manual Trip and RTCB testing on the closed breakers cannot be performed without causing a trip.~~]~~ 3

REFERENCES

~~1. 10 CFR 50, Appendix A.~~

1 2. 10 CFR ~~10~~0. 50.67

2 ~~3~~. FSAR, Section ~~[~~7.2~~]~~. U

~~4. NRC Safety Evaluation Report, [Date].~~

~~5. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989.~~

1 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.2 BASES, REACTOR PROTECTION SYSTEM (RPS) LOGIC AND TRIP INITIATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The heading for ISTS 3.3.3 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2). Therefore, ISTS 3.3.2 is not included in the PSL ITS. The Reactor Protective System (RPS) Logic and Trip Initiation (ISTS 3.3.3) is renumbered as ITS 3.3.2.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Figure B 3.3.1-1 was removed from the ITS 3.3.1-1 Bases because the figure is included in the UFSAR.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.2, REACTOR PROTECTION SYSTEM (RPS) LOGIC AND TRIP INITIATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

3.3.3, ESFAS Instrumentation

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.3
Table 3.3.3-1

~~3.3.2.1~~ The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE ~~with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4~~.

LA01

LCO 3.3.3
Table 3.3.3-1

APPLICABILITY: As shown in Table 3.3-3.

Add proposed ITS 3.3.3 ACTIONS Note

A02

ACTION:

Table 3.3.3-1

~~a.~~ With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, ~~declare the channel inoperable and~~ apply the applicable ACTION requirement of Table 3.3-3 ~~until the channel is restored to OPERABLE status~~ with the trip setpoint adjusted consistent with the Trip Setpoint value.

ACTIONS for associated
* Table 3.3.3-1 Functions 1-6

Table 3.3.3-1
Note (b)

ACTIONS for associated
* Table 3.3.3-1 Functions 1-6

~~b.~~ With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

~~4.3.2.1.1~~ Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes ~~and at the frequencies~~ shown in Table 4.3-2.

SR 3.3.3.1, SR 3.3.3.4,
SR 3.3.3.2
Table 3.3.3-1

In accordance with the Surveillance Frequency Control Program

~~4.3.2.1.2~~ The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. ~~The total bypass function shall be demonstrated OPERABLE~~ in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing ~~of each channel affected by bypass operation~~.

Table 3.3.3-1
SR 3.3.3.3

SR 3.3.3.4

include the bypass removal functions

A03

~~4.3.2.1.3~~ The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program. ~~Each test shall include at least one channel per function.~~

Table 3.3.3-1
SR 3.3.3.5

A03

ACTIONS for associated
*Table 3.3.3-1 Functions 1-6
ACTION A,E,I – CSAS
ACTION B,F,I – RAS
ACTION C,E,I – AFAS
ACTION C,E,I – AUX FW ISOL
ACTION D,F,I – SIAS, CIAS, MSIS
ACTION G,H,I – BYPASSES

A01

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LA02

A04

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ REQUIRED CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
1. SAFETY INJECTION (SIAS)						
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8	See ITS 3.3.4
Table 3.3.3-1 1.a b. Containment Pressure – High	~~4~~	~~2~~	~~3~~ 4	1, 2, 3	9	A04
Table 3.3.3-1 1.b c. Pressurizer Pressure – Low	~~4~~	~~2~~	~~3~~ 4	1, 2, 3(a)	9	A04
2. CONTAINMENT SPRAY (CSAS)						
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8	See ITS 3.3.4
Table 3.3.3-1 2.a b. Containment Pressure – High-High	~~4~~	~~2~~(b)	~~3~~ 4	1, 2, 3	10A, 10B	A04
Table 3.3.3-1 Note (d) 3. CONTAINMENT ISOLATION (CIS)						
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8	See ITS 3.3.4
Table 3.3.3-1 3.a b. Containment Pressure – High	~~4~~	~~2~~	~~3~~ 4	1, 2, 3	9	A04
Table 3.3.3-1 3.b c. Containment Radiation – High	~~4~~	~~2~~	~~3~~ 4	1, 2, 3, 4	9	A04
d. SIAS	---------- (See Functional Unit 1 above) ----------					
4. MAIN STEAM LINE ISOLATION (MSIS)						
a. Manual (Trip Buttons)	2/steam generator	1/steam generator	2/operating steam generator	1, 2, 3, 4	8	See ITS 3.3.4
Table 3.3.3-1 4.a b. Steam Generator Pressure – Low	~~4/steam generator~~	~~2/steam generator~~	~~3~~/steam 4 generator	1, 2, 3(c)	9	A04

Add Footnote (f)

L01

A01

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LA02

A04

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ REQUIRED CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
5. CONTAINMENT SUMP RECIRCULATION (RAS)						
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	8	See ITS 3.3.4
Table 3.3.3-1 5.a b. Refueling Water Tank - Low	~~4~~	~~2~~	~~3~~ 4	1, 2, 3	13	A04
6. LOSS OF POWER						See ITS 3.3.5
a. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12	
b. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12	
c. 480 V Emergency Bus Under-voltage (Degraded Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12	
7. AUXILIARY FEEDWATER (AFAS)						
a. Manual (Trip Buttons)	4/SG	2/SG	4/SG	1, 2, 3	11	See ITS 3.3.4
b. Automatic Actuation Logic	4/SG	2/SG	3/SG	1, 2, 3	11	
Table 3.3.3-1 6.a. 6.b c. SG Level (1A/1B) - Low	~~4/SG~~	~~2/SG~~	~~3~~/SG	1, 2, 3	14a, 14b, 15	A04
8. AUXILIARY FEEDWATER ISOLATION						
Table 3.3.3-1 6.c a. SG 1A – SG 1B Differential Pressure	~~4/SG~~	~~2/SG~~	~~3~~/SG	1, 2, 3	14a, 14b, 15	A04
Table 3.3.3-1 6.d b. Feedwater Header 1A – 1B Differential Pressure	~~4/SG~~	~~2/SG~~	~~3~~/SG	1, 2, 3	14a, 15	A04

4

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

M01 — Add Table 3.3.3-1 Footnote (a)

M02 — Add Table 3.3.3-1 Footnote (b)

Pressurizer Pressure - Low; manually

Table 3.3.3-1 Note (c)

(a) ~~Trip function~~ may be bypassed ~~in this MODE~~ when pressurizer pressure is < 1725 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1725 psia.

also required as a permissive to initiate containment spray

Table 3.3.3-1 Note (d)

(b) ~~An~~ SIAS ~~signal~~ is ~~first necessary to enable CSAS logic~~.

Steam Generator Pressure - Low; manually; when steam generator pressure is <

Table 3.3.3-1 Note (e)

(c) ~~Trip function~~ may be bypassed ~~in this MODE below~~ 685 psig; bypass shall be automatically removed ~~at or above~~ 685 psig.

L01 — Add Table 3.3.3-1 Footnote (f)

whenever steam generator pressure is ≥

A02 — Add proposed LCO 3.3.3 ACTIONS Note

ACTION STATEMENTS

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.3.4

ACTION 9 - ~~With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:~~

ACTION D (Condition)

One or more Functions with one ESFAS trip unit or associated instrument channel (except CSAS, RAS or AFAS).

Place affected trip unit

ACTION D.1

~~a.~~ ~~The inoperable channel is placed~~ in ~~either the~~ bypass~~ed~~ or trip~~ped condition~~ within 1 hour. ~~For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to~~ 48 hours ~~from time of initial loss of OPERABILITY; however, the inoperable~~ channel ~~shall then be either~~ restore~~d~~ to OPERABLE status or place~~d~~ in ~~the~~ trip~~ped condition~~.

ACTION D.2.1
ACTION D.2.2

channel; affected trip unit; within

~~b.~~ ~~Within one hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.~~

A05

ACTION F

~~c.~~ ~~The Minimum Channels OPERABLE requirement is met; however,~~ one ~~additional~~ channel may be bypassed for up to 48 hours ~~while performing tests and maintenance on that channel~~ provided the other inoperable channel is placed in the tripped condition.

L02

within 1 hour

With one or more Functions with two ESFAS trip unit or associated instrument channel (except CSAS or AFAS),

A01

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION 10A - ~~With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:~~

ACTION D (Condition) — One or more Functions with one ESFAS trip unit or associated instrument channel (except CSAS, RAS or AFAS).

Place affected trip unit

ACTION D.1 — ~~a.~~ ~~The inoperable channel is placed~~ in ~~the~~ bypass~~ed~~ or trip~~ped~~ ~~condition and the Minimum Channels OPERABLE requirement is demonstrated~~ within 1 hour. ~~If the inoperable channel can not be~~ restore~~d~~ to OPERABLE status within 48 hours, ~~then~~ place ~~the inoperable channel~~ in ~~the~~ trip~~ped condition~~.

ACTION D.2.1 / ACTION D.2.2 — channel; affected trip unit; or

~~b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.~~ A05

ACTION E (Condition) — Two CSAS trip units or associated instrument channels inoperable.

ACTION 10B - ~~With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been~~ bypass~~ed~~ and the other ~~inoperable~~ ~~channel has been placed~~ in ~~the~~ trip~~ped~~ ~~condition~~ within 1 hour. Restore one ~~of the inoperable channels~~ to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at~~ least ~~HOT STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours.

ACTION E.1 — Place one trip unit in; place; trip unit

ACTION E.2 — trip unit

ACTION I — MODE 3; MODE 4; 12

ACTION 11 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.4

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

See ITS 3.3.5

A01

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION B (Condition)

ACTION 13 - ~~With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:~~

One Containment Sump Recirculation Actuation Signal (RAS) trip unit or associated instrument channel inoperable.

trip unit to OPERABLE status

Place affected trip unit

ACTION B.1
ACTION B.2

~~a.~~ ~~The inoperable channel is placed~~ in ~~either the~~ bypass~~ed~~ or trip~~ped condition~~ within 1 hour. ~~If OPERABILITY cannot be~~ restore~~d~~ within 48 hours or in accordance with the Risk Informed Completion Time Program, be in ~~at least~~ ~~HOT STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours.

MODE 3

MODE 4

12

ACTION I

ACTION F

With one or more Functions with two ESFAS trip unit or associated instrument channel (except CSAS or AFAS),

~~The Minimum Channels OPERABLE requirement is met; however,~~ one ~~additional~~ channel may be bypassed ~~for up to 2 hours while performing tests and maintenance on that channel~~ provided the other inoperable channel is placed in the tripped condition.

within 1 hour

L02

ACTION C NOTE (Condition)

NOTE Required Action C.2.2.2 must be completed whenever this Condition is entered.

A07

ACTION 14 - ~~With the number of channels OPERABLE one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:~~

ACTION C (Condition)

One or more Functions with one Auxiliary Feedwater Actuation Signal (AFAS) trip unit or associated instrument channel inoperable.

Place affected trip unit

ACTION C.1

~~a.~~ ~~The inoperable channel is placed~~ in ~~either the~~ bypass~~ed~~ or trip~~ped condition~~ within 1 hour. ~~If an inoperable SG level channel can not be~~ restore~~d~~ to OPERABLE status within 48 hours, ~~then AFAS-1 or AFAS-2 as applicable in the inoperable channel shall be~~ place~~d~~ in ~~the~~ bypass~~ed condition~~. ~~If an inoperable SG DP or FW Header DP channel can not be~~ restore~~d~~ to OPERABLE status within 48 hours, ~~then both AFAS-1 and AFAS-2 in the inoperable channel shall be~~ place~~d~~ in ~~the~~ bypass~~ed condition~~. ~~The~~ channel ~~shall be returned~~ to OPERABLE status ~~no later than during the next COLD SHUTDOWN~~.

ACTION C.2.1
ACTION C.2.2.1

trip unit

affected trip unit

within

LA03

ACTION C.2.1
ACTION C.2.2.1

trip unit

affected trip unit

LA03

ACTION C.2.2.2

Restore

Prior to entering MODE 3 following next MODE 5 entry

A07

~~b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.~~

A05

ACTION E (Condition)

One or more Functions with two AFAS trip units or associated instrument channels inoperable.

one trip unit

ACTION 15 - ~~With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been~~ bypass~~ed~~ and ~~the other inoperable channel has been~~ place~~d~~ in ~~the~~ trip~~ped condition~~ within 1 hour. Restore one ~~of the inoperable channels~~ to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least~~ ~~HOT STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours.

Place one trip unit in

ACTION E.1

trip unit

ACTION E.2

MODE 3

MODE 4

12

ACTION I

Add proposed ACTIONS G and H

L03

Add proposed ACTION I for SIAS, CIS, MSIS, and RAS

A06

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

LA01

FUNCTIONAL UNIT	~~TRIP SETPOINT~~	ALLOWABLE VALUES	
1. SAFETY INJECTION (SIAS)			
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
Table 3.3.3-1 1.a b. Containment Pressure – High	~~≤ 5 psig~~	≤ 5 psig	
Table 3.3.3-1 1.b c. Pressurizer Pressure – Low	~~≥ 1600 psia~~	≥ 1600 psia	
2. CONTAINMENT SPRAY (CSAS)			
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
Table 3.3.3-1 2.a b. Containment Pressure – High-High	~~≤ 10 psig~~	≤ 10 psig	
3. CONTAINMENT ISOLATION (CIS)			
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
Table 3.3.3-1 3.a b. Containment Pressure – High	~~≤ 5 psig~~	≤ 5 psig	
Table 3.3.3-1 3.b c. Containment Radiation – High	~~≤ 10 R/hr~~	≤ 10 R/hr	
d. SIAS	----------(See FUNCTIONAL UNIT 1 above)----------		
4. MAIN STEAM LINE ISOLATION (MSIS)			
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
Table 3.3.3-1 4.a b. Steam Generator Pressure – Low	~~≥ 585 psig~~	≥ 585 psig	
5. CONTAINMENT SUMP RECIRCULATION (RAS)			
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
Table 3.3.3-1 5.a b. Refueling Water Tank – Low	~~48 inches above tank bottom~~	48 inches above tank bottom	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

LA01

FUNCTIONAL UNIT	~~TRIP VALUE~~	ALLOWABLE VALUES	
6. LOSS OF POWER			See ITS 3.3.5
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	≥ 2900 volts with a 1 ± .5 second time delay	≥ 2900 volts with a 1 ± .5 second time delay	
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3831 volts with a 18 ± 2 second time delay	≥ 3831 volts with a 18 ± 2 second time delay	
c. 480 volts Emergency Bus Undervoltage (Degraded Voltage)	≥ 415 volts with a ≤ 9 second time delay	≥ 415 volts with a ≤ 9 second time delay	
7. AUXILIARY FEEDWATER (AFAS)			
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
b. Automatic Actuation Logic	Not Applicable	Not Applicable	
Table 3.3.3-1 6.a, 6.b c. SG 1A & 1B Level Low	~~≥ 19.0%~~	≥ 18.0 %	
8. AUXILIARY FEEDWATER ISOLATION			
Table 3.3.3-1 6.c a. Steam Generator ΔP – High	~~≤ 275 psid~~	89.2 to 281 psid	
Table 3.3.3-1 6.d b. Feedwater Header High ΔP	~~≤ 150.0 psid~~	56.0 to 157.5 psid	

A01

~~DELETED~~

A01

~~DELETED~~

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	SR 3.3.3.1 CHANNEL CHECK	SR 3.3.3.4 CHANNEL CALIBRATION	SR 3.3.3.2 CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED	
SR 3.3.3.3 SR 3.3.3.5 1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.	See ITS 3.3.4
b. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3	
c. Pressurizer Pressure – Low	SFCP	SFCP	SFCP	1, 2, 3	
d. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3	See ITS 3.3.4
SR 3.3.3.5 2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.	See ITS 3.3.4
b. Containment Pressure – ~~–~~ High-High	SFCP	SFCP	SFCP	1, 2, 3	
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3	See ITS 3.3.4
SR 3.3.3.5 3. CONTAINMENT ISOLATION (CIS)					
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.	See ITS 3.3.4
b. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3	
c. Containment Radiation – High	SFCP	SFCP	SFCP	1, 2, 3, 4	
d. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3	See ITS 3.3.4
e. SIAS	N.A.	N.A.	SFCP	N.A.	
SR 3.3.3.3 SR 3.3.3.5 4. MAIN STEAM LINE ISOLATION (MSIS)					
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.	See ITS 3.3.4
b. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2, 3	
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3	See ITS 3.3.4
SR 3.3.3.5 5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	N.A.	N.A.	SFCP	N.A.	See ITS 3.3.4
b. Refueling Water Storage Tank – Low	SFCP	SFCP	SFCP	1, 2, 3	
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3	See ITS 3.3.4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

SR 3.3.3.1 SR 3.3.3.4 SR 3.3.3.2

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED	
6. LOSS OF POWER					See ITS 3.3.5
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	SFCP	SFCP	SFCP	1, 2, 3	
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3	
c. 480 V Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3	
SR 3.3.3.5 7. AUXILIARY FEEDWATER (AFAS)					
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	1, 2, 3	See ITS 3.3.4
b. SG Level (A/B) – Low	SFCP	SFCP	SFCP	1, 2, 3	
c. Automatic Actuation Logic	N.A.	N.A.	SFCP	1, 2, 3	See ITS 3.3.4
SR 3.3.3.5 8. AUXILIARY FEEDWATER ISOLATION					
a. SG Level (A/B) – Low and SG Differential Pressure (BtoA/AtoB) – High	N.A.	SFCP	SFCP	1, 2, 3	
b. SG Level (A/B) – Low and Feedwater Header Differential Pressure (BtoA/AtoB) – High	N.A.	SFCP	SFCP	1, 2, 3	

A01

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) The logic circuits shall be tested manually in accordance with the Surveillance Frequency Control Program.

See ITS 3.3.4

A01

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.3
Table 3.3.3-1

~~3.3.2~~ The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE ~~with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4~~.

LA01

LCO 3.3.3
Table 3.3.3-1

APPLICABILITY: As shown in Table 3.3-3.

Add proposed ITS 3.3.3 ACTIONS Note

A02

ACTION:

Table 3.3.3-1
ACTIONS for
* Table 3.3.3-1 Functions 1-6
Table 3.3.3-1 Note (b)

~~a.~~ With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, ~~declare the channel inoperable and~~ apply the applicable ACTION requirement of Table 3.3-3 ~~until the channel is restored to OPERABLE status~~ with the trip setpoint adjusted consistent with the Trip Setpoint value.

ACTIONS for
* Table 3.3.3-1 Functions 1-6

~~b.~~ With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

~~4.3.2.1~~ Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES ~~and at the frequencies~~ shown in Table 4.3-2.

SR 3.3.3.1, SR 3.3.3.4,
SR 3.3.3.2
Table 3.3.3-1

In accordance with the Surveillance Frequency Control Program

~~4.3.2.2~~ The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. ~~The total bypass function shall be demonstrated OPERABLE~~ in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing ~~of each channel affected by bypass operation~~.

Table 3.3.3-1
SR 3.3.3.3

SR 3.3.3.4

A03

include the bypass removal functions

~~4.3.2.3~~ The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program. ~~Each test shall include at least one channel per function.~~

Table 3.3.3-1
SR 3.3.3.5

A03

ACTIONS for associated
*Table 3.3.3-1 Functions 1-6
ACTION A,E,I – CSAS
ACTION B,F,I – RAS
ACTION C,E,I – AFAS
ACTION C,E,I – AUX FW ISOL
ACTION D,F,I – SIAS, CIAS, MSIS
ACTION G,H,I – BYPASSES

A01

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LA02

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ REQUIRED CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
1. SAFETY INJECTION (SIAS)						
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	12	See ITS 3.3.4
Table 3.3.3-1 1.a b. Containment Pressure – High	~~4~~	~~2~~	~~3~~ 4	1, 2, 3	13, 14	A04
Table 3.3.3-1 1.b c. Pressurizer Pressure – Low	~~4~~	~~2~~	~~3~~ 4	1, 2, 3(a)	13, 14	A04
d. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	12	See ITS 3.3.4
2. CONTAINMENT SPRAY (CSAS)						
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	12	See ITS 3.3.4
Table 3.3.3-1 2.a b. Containment Pressure – High-High	~~4~~	~~2~~	~~3~~ 4	1(b), 2(b), 3(b)	18A, 18B	A04
c. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	12	See ITS 3.3.4
3. CONTAINMENT ISOLATION (CIAS)						
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12	See ITS 3.3.4
b. Safety Injection (SIAS)	See Functional Unit 1 for all Safety Injection Initiating Functions and Requirements					
Table 3.3.3-1 3.a c. Containment Pressure – High	~~4~~	~~2~~	~~3~~ 4	1, 2, 3	13, 14	A04
Table 3.3.3-1 3.b d. Containment Radiation – High	~~4~~	~~2~~	~~3~~ 4	1, 2, 3	13, 14	A04
e. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	12	See ITS 3.3.4

A01

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LA02

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ REQUIRED CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
4. MAIN STEAM LINE ISOLATION (MSIS)						
a. Manual (Trip Buttons)	2	1	2	1, 2, 3	16	See ITS 3.3.4
Table 3.3.3-1 4.a — b. Steam Generator Pressure – Low	~~4/steam generator~~	~~2/steam generator~~	~~3~~/steam generator (4)	1, 2, 3(c)	13, 14	A04
Table 3.3.3-1 4.b — c. Containment Pressure – High	~~4~~	~~2~~	~~3~~ (4)	1, 2, 3	13, 14	A04
d. Automatic Actuation Logic	2	1	2	1, 2, 3	12	See ITS 3.3.4
5. CONTAINMENT SUMP RECIRCULATION (RAS)						
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12	See ITS 3.3.4
Table 3.3.3-1 5.a — b. Refueling Water Tank - Low	~~4~~	~~2~~	~~3~~ (4)	1, 2, 3	19	A04
c. Automatic Actuation Logic	2	1	2	1, 2, 3	12	See ITS 3.3.4

A04

Add Footnote (f)

L01

A01

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LA02

A04

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ CHANNELS OPERABLE (REQUIRED)	APPLICABLE MODES	ACTION	
6. LOSS OF POWER (LOV)						See ITS 3.3.5
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	17A	
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17B	
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17B	
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17B	
7. AUXILIARY FEEDWATER (AFAS)						
a. Manual (Trip Buttons)	4/SG	2/SG	4/SG	1, 2, 3	15	See ITS 3.3.4
b. Automatic Actuation Logic	4/SG	2/SG	3/SG	1, 2, 3	15	
c. SG Level (2A/2B) – Low	~~4/SG~~	~~2/SG~~	~~3~~/SG	1, 2, 3	20a, 20b, 21	A04
8. AUXILIARY FEEDWATER ISOLATION						
a. SG 2A – SG 2B Differential Pressure	~~4/SG~~	~~2/SG~~	~~3~~/SG	1, 2, 3	20a, 20b, 21	A04
b. Feedwater Header 2A – 2B Differential Pressure	~~4/SG~~	~~2/SG~~	~~3~~/SG	1, 2, 3	20a, 21	A04

Table 3.3.3-1 6.a, 6.b

Table 3.3.3-1 6.c

Table 3.3.3-1 6.d

4

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

Pressurizer Pressure - Low | manually

Table 3.3.3-1 Note (c)

(a) ~~Trip function~~ may be bypassed ~~in this MODE~~ when pressurizer pressure is less than 1836 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to 1836 psia.

also required as a permissive to initiate containment spray

Table 3.3.3-1 Note (d)

(b) ~~An~~ SIAS ~~signal~~ is ~~first necessary to enable CSAS logic~~.

Steam Generator Pressure - Low | manually | when steam generator pressure is <

Table 3.3.3-1 Note (e)

(c) ~~Trip function~~ may be bypassed ~~in this MODE~~ ~~below~~ 700 psia; bypass shall be automatically removed ~~at or above~~ 700 psia.

whenever steam generator pressure is ≥

Add proposed LCO 3.3.3 ACTIONS Note — L01

ACTION OF STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.3.4

ACTION D (Condition) NOTE: NOTE Required Action D.2 must be completed whenever this Condition is entered.

A07

One or more Functions with one ESFAS trip unit or associated instrument channel (except CSAS, RAS or AFAS).

ACTION D (Condition)

ACTION 13 - ~~With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed~~ in ~~the~~ bypass~~ed~~ or trip~~ped condition~~ within 1 hour.

ACTION D.1: Place affected trip unit

ACTION D.2: Restore

~~The~~ channel ~~shall be returned~~ to OPERABLE status ~~no later than during the next COLD SHUTDOWN~~.

Prior to entering MODE 3 following next MODE 5 entry

A07

~~With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.~~

A05

~~Process Measurement Circuit~~	~~Functional Unit Bypassed~~
~~1. Containment Pressure -~~	~~Containment Pressure – High (SIAS, CIAS, CSAS)~~ ~~Containment Pressure – High (RPS)~~
~~2. Steam Generator Pressure -~~	~~Steam Generator Pressure – Low (MSIS)~~ ~~AFAS-1 and AFAS-2 (AFAS)~~ ~~Thermal Margin/Low Pressure (RPS)~~ ~~Steam Generator Pressure – Low (RPS)~~
~~3. Steam Generator Level -~~	~~Steam Generator Level – Low (RPS)~~ ~~If SG-2A, then AFAS-1 (AFAS)~~ ~~If SG-2B, then AFAS-2 (AFAS)~~
~~4. Pressurizer Pressure -~~	~~Pressurizer Pressure – High (RPS)~~ ~~Pressurizer Pressure – Low (SIAS)~~ ~~Thermal Margin/Low Pressure (RPS)~~

LA04

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION F (Condition)

ACTION F.1

ACTION 14 - ~~With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:~~

One or more Functions with two ESFAS trip unit or associated instrument channel (except CSAS or AFAS).

Place one trip unit in

~~a.~~ ~~Verify that one of the inoperable channels has been~~ bypass~~ed~~ and place the other ~~inoperable channel~~ in ~~the~~ trip~~ped condition~~ within 1 hour.

trip unit

Add proposed ACTION F.2

L02

~~b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below.~~

A04

LA04

~~Process Measurement Circuit~~	~~Functional Unit Bypassed/Tripped~~
~~1. Containment Pressure -~~	~~Containment Pressure -- High (SIAS, CIAS, CSAS)~~ ~~Containment Pressure -- High (RPS)~~
~~2. Steam Generator Pressure -~~	~~Steam Generator Pressure -- Low (MSIS)~~ ~~AFAS-1 and AFAS-2 (AFAS)~~ ~~Thermal Margin/Low Pressure (RPS)~~ ~~Steam Generator Pressure -- Low (RPS)~~
~~3. Steam Generator Level -~~	~~Steam Generator Level -- Low (RPS)~~ ~~If SG-2A, then AFAS-1 (AFAS)~~ ~~If SG-2B, then AFAS-2 (AFAS)~~
~~4. Pressurizer Pressure -~~	~~Pressurizer Pressure -- High (RPS)~~ ~~Pressurizer Pressure -- Low (SIAS)~~ ~~Thermal Margin/Low Pressure (RPS)~~

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.4

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 17A - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or place the inoperable channel in the tripped condition and verify that the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

See ITS 3.3.5

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 17B - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or place the inoperable channel in the tripped condition and verify that the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

See ITS 3.3.5

ACTION 18A - ~~With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:~~

ACTION D (Condition)

One or more Functions with one ESFAS trip unit or associated instrument channel (except CSAS, RAS or AFAS).

Place affected trip unit

ACTION D.1

~~a.~~ ~~The inoperable channel is placed~~ in ~~either the~~ bypass~~ed~~ or trip~~ped~~ ~~condition and the Minimum Channels OPERABLE requirement is demonstrated~~ within 1 hour. ~~If the inoperable channel can not be~~ restore~~d~~ to OPERABLE status within 48 hours, ~~then~~ place ~~the inoperable channel~~ in ~~the~~ trip~~ped~~ ~~condition~~.

ACTION D.2.1
ACTION D.2.2

or

affected trip unit

channel

~~b. With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.~~

A05

Two CSAS trip units or associated instrument channels inoperable.

ACTION E (Condition)

ACTION 18B - ~~With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been~~ bypass~~ed~~ and the other ~~inoperable channel has been placed~~ in ~~the~~ trip~~ped~~ ~~condition~~ within 1 hour. Restore one ~~of the inoperable channels~~ to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ 6 hours.

ACTION E.1

Place one trip unit in

place

trip unit

ACTION E.2

ACTION I

MODE 3

MODE 4

12

ACTION B (Condition)

ACTION 19 - ~~With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:~~

One Containment Sump Recirculation Actuation Signal (RAS) trip unit or associated instrument channel inoperable.

place affected trip unit

trip unit to OPERABLE status

ACTION B.1
ACTION B.2

a. Within 1 hour ~~the inoperable channel is placed~~ in ~~either the~~ bypass~~ed~~ or trip~~ped~~ ~~condition~~. ~~If OPERABILITY cannot be~~ restore~~d~~ within 48 hours or in accordance with the Risk Informed Completion Time Program, be in ~~at least HOT STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following~~ 6 hours.

ACTION I

MODE 3

MODE 4

12

ACTION F

With one or more Functions with two ESFAS trip unit or associated instrument channel (except CSAS or AFAS).

~~The Minimum Channels OPERABLE requirement is met; however,~~ one ~~additional~~ channel may be bypassed ~~for up to 2 hours for surveillance testing per Specification 4.3.2.1~~.

within 1 hour

L02

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION C NOTE (Condition)

NOTE Required Action C.2.2.2 must be completed whenever this Condition is entered.

A07

ACTION 20 - ~~With the number of channels OPERABLE one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:~~

ACTION C (Condition)

One or more Functions with one Auxiliary Feedwater Actuation Signal (AFAS) trip unit or associated instrument channel inoperable.

Place affected trip unit

trip unit

~~a.~~ ~~The inoperable channel is placed~~ in ~~either the~~ bypass~~ed~~ or trip~~ped ped condition~~ within 1 hour. ~~If an inoperable SG level channel can not be~~ restore~~d~~ to OPERABLE status within 48 hours, ~~then AFAS-1 or AFAS-2 as applicable in the inoperable channel shall be~~ place~~d~~ in ~~the~~ bypass~~ed condition~~.

ACTION C.1

ACTION C.2.1

ACTION

affected trip unit

within

LA03

~~If an inoperable SG DP or FW Header DP channel can not be~~ restore~~d~~ to OPERABLE status within 48 hours, ~~then both AFAS-1 and AFAS-2 in the inoperable channel shall be~~ place~~d~~ in ~~the~~ bypass~~ed condition~~. ~~The~~ channel ~~shall be returned~~ to OPERABLE status ~~no later than during the next COLD SHUTDOWN~~.

trip unit

LA03

ACTION C.2.1

ACTION

affected trip unit

ACTION C.2.2.2

Prior to entering MODE 3 following next MODE 5 entry

Restore

A07

~~b~~ ~~With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.~~

A05

One or more Functions with two AFAS trip units or associated instrument channels inoperable.

one trip unit

ACTION E (Condition)

ACTION 21 - ~~With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been~~ bypass~~ed~~ and ~~the other inoperable channel~~ place~~d~~ in ~~the~~ trip~~ped condition~~ within 1 hour. Restore one ~~of the inoperable channels~~ to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours.

Place one trip unit in

ACTION E.1

trip unit

ACTION E.2

ACTION I

MODE 3

MODE 4

12

Add proposed ACTIONS G and H

L03

Add proposed ACTION I for SIAS, CIS, MSIS, CSAS, and AFAS

A06

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

LA01

FUNCTIONAL UNIT	~~TRIP SETPOINT~~	ALLOWABLE VALUES	
1. SAFETY INJECTION (SIAS)			
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
Table 3.3.3-1 1.a b. Containment Pressure – High	~~≤ 3.5 psig~~	≤ 3.6 psig	
Table 3.3.3-1 1.b c. Pressurizer Pressure – Low	~~≥ 1736 psia~~	≥ 1728 psia	
d. Automatic Actuation Logic	Not Applicable	Not Applicable	See ITS 3.3.4
2. CONTAINMENT SPRAY (CSAS)			
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
Table 3.3.3-1 2.a b. Containment Pressure – High-High	~~≤ 5.40 psig~~	≤ 5.50 psig	
c. Automatic Actuation Logic	Not Applicable	Not Applicable	See ITS 3.3.4
3. CONTAINMENT ISOLATION (CIAS)			
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
b. Safety Injection (SIAS)	Not Applicable	Not Applicable	
Table 3.3.3-1 3.a c. Containment Pressure – High	~~≤ 3.5 psig~~	≤ 3.6 psig	
Table 3.3.3-1 3.b d. Containment Radiation – High	~~≤ 10 R/hr~~	≤ 10 R/hr	
e. Automatic Actuation Logic	Not Applicable	Not Applicable	See ITS 3.3.4
4. MAIN STEAM LINE ISOLATION			
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
Table 3.3.3-1 4.a b. Steam Generator Pressure – Low	~~≥ 600 psia~~	≥ 567 psia	
Table 3.3.3-1 4.b c. Containment Pressure – High	~~≤ 3.5 psig~~	≤ 3.6 psig	
d. Automatic Actuation Logic	Not Applicable	Not Applicable	See ITS 3.3.4

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

LA01

FUNCTIONAL UNIT	~~TRIP VALUE~~	ALLOWABLE VALUES	
5. CONTAINMENT SUMP RECIRCULATION (RAS)			
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
Table 3.3.3-1 5.a — b. Refueling Water Tank – Low	~~5.67 feet above tank bottom~~	4.62 feet to 6.24 feet above tank bottom	
c. Automatic Actuation Logic	Not Applicable	Not Applicable	See ITS 3.3.4
6. LOSS OF POWER			
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ 3120 volts	≥ 3120 volts	See ITS 3.3.5
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	≥ 360 volts	≥ 360 volts	
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	≥ 3848 volts with < 10-second time delay	≥ 3848 volts with < 10-second time delay	
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	≥ 432 volts	≥ 432 volts	
7. AUXILIARY FEEDWATER (AFAS)			
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
b. Automatic Actuation Logic	Not Applicable	Not Applicable	
Table 3.3.3-1 6.a, 6.b — c. SG 2A & 2B Level Low	~~≥ 19.0%~~	≥ 18.0 %	
8. AUXILIARY FEEDWATER ISOLATION			
Table 3.3.3-1 6.c — a. Steam Generator ΔP – High	~~≤ 275 psid~~	89.2 to 281 psid	
Table 3.3.3-1 6.d — b. Feedwater Header ΔP – High	~~≤ 150.0 psid~~	56.0 to 157.5 psid	

A01

~~DELETED~~

A01

~~DELETED~~

A01

~~DELETED~~

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	SR 3.3.3.1 CHANNEL CHECK	SR 3.3.3.4 CHANNEL CALIBRATION	SR 3.3.3.2 CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
SR 3.3.3.3 SR 3.3.3.5 1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	1, 2, 3, 4	See ITS 3.3.4
b. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3	
c. Pressurizer Pressure – Low	SFCP	SFCP	SFCP	1, 2, 3	
d. Automatic Actuation Logic	N.A.	N.A.	SFCP(1), SFCP(3)	1, 2, 3, 4	See ITS 3.3.4
SR 3.3.3.5 2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	1, 2, 3, 4	See ITS 3.3.4
b. Containment Pressure – High-High	SFCP	SFCP	SFCP	1, 2, 3	
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1), SFCP(3)	1, 2, 3, 4	See ITS 3.3.4
SR 3.3.3.5 3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	SFCP	1, 2, 3, 4	See ITS 3.3.4
b. Safety Injection SIAS	N.A.	N.A.	SFCP	1, 2, 3, 4	
c. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3	
d. Containment Radiation – High	SFCP	SFCP	SFCP	1, 2, 3	
e. Automatic Actuation Logic	N.A.	N.A.	SFCP(1), SFCP(3)	1, 2, 3, 4	See ITS 3.3.4
SR 3.3.3.3 SR 3.3.3.5 4. MAIN STEAM LINE ISOLATION					
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	1, 2, 3	See ITS 3.3.4
b. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2, 3	
c. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3	
d. Automatic Actuation Logic	N.A.	N.A.	SFCP(1), SFCP(3)	1, 2, 3, 4	See ITS 3.3.4
SR 3.3.3.5 5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	N.A.	N.A.	SFCP	N.A.	See ITS 3.3.4
b. Refueling Water Tank – Low	SFCP	SFCP	SFCP	1, 2, 3	
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1), SFCP(3)	1, 2, 3	See ITS 3.3.4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	SR 3.3.3.1 CHANNEL CHECK	SR 3.3.3.4 CHANNEL CALIBRATION	SR 3.3.3.2 CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. LOSS OF POWER (LOV)				See ITS 3.3.5
a. 4.16 kV and 480 V Emergency Bus Undervoltage (Loss of Voltage)	SFCP	SFCP	SFCP	1, 2, 3, 4
b. 4.16 kV and 480 V Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3, 4
SR 3.3.3.5 7. AUXILIARY FEEDWATER (AFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	1, 2, 3 — See ITS 3.3.4
b. SG Level (A/B) – Low	SFCP	SFCP	SFCP	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1), SFCP(2)	1, 2, 3 — See ITS 3.3.4
SR 3.3.3.5 8. AUXILIARY FEEDWATER ISOLATION				
a. SG Level (A/B) – Low and SG Differential Pressure (B to A/A to B) – High	N.A.	SFCP	SFCP	1, 2, 3
b. SG Level (A/B) – Low and Feedwater Header Differential Pressure (B to A/A to B) – High	N.A.	SFCP	SFCP	1, 2, 3

TABLE NOTATION

(1) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay (solid-state component) and verification of the OPERABILITY of each initiation relay (solid-state component).

(2) An actuation relay test shall be performed which shall include the energization/de-energization of each actuation relay and verification of the OPERABILITY of each actuation relay.

(3) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Testing of the ESFAS subgroup relays shall be performed in accordance with the Surveillance Frequency Control Program.

See ITS 3.3.4

DISCUSSION OF CHANGES
ITS 3.3.3, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS Table 3.3-3 Actions describe the actions to be taken when ESFAS Function instrument channels are inoperable. ITS 3.3.3 also describes Actions to be taken when ESFAS Function instrument channels are inoperable and contains a note that separate condition entry is allowed for each ESFAS Function. This changes the CTS by adding a Note stating that separate condition entry is allowed for each Function, those ESFAS Functions being the SIAS, CSAS, CIS (Unit 1) and CIAS (Unit 2), MSIS, RAS, and AFAS Functions.

The purpose of the CTS Actions is to provide the appropriate compensatory actions for inoperable ESFAS Functions. This proposed change will allow separate condition entry for each ESFAS Function. The Note clarifies that ESFAS Functions are treated as separate entities, each with separate Completion Times. In addition, the AFAS Functions are allowed separate Condition entry on a per steam generator basis since the channels associated with each steam generator will provide the associated AFAS trip based on the logic associated with the channels on a steam generator basis. This change is acceptable because it clearly states the current requirement. The CTS considers each ESFAS Function to be separate and independent from the others. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 Unit 1 CTS 4.3.2.1.2 and Unit 2 CTS 4.3.2.2 require the logic for the bypasses to be demonstrated OPERABLE during the at-power CHANNEL FUNCTIONAL TEST. These Surveillance Requirements also require, “The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.” Unit 1 CTS 4.3.2.1.3 and Unit 2 CTS 4.3.2.3 require ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME testing for each ESFAS function and includes a statement that each test shall include at least one channel per function. ITS SR 3.3.3.3 requires a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function. ITS SR 3.3.3.4 requires a CHANNEL CALIBRATION of each ESFAS instrument channel, including bypass removal functions. These ITS Surveillances do not explicitly state that the total bypass function shall be demonstrated OPERABLE during the Channel Calibration test of each channel affected by bypass operation or that the ESF RESPONSE TIME test must include at least one channel per function.

The purpose of the CTS statements in the Surveillance Requirements associated with the CHANNEL FUNCTIONAL TEST of the bypass channels and the ESF RESPONSE TIME test is to ensure required features are tested to verify the ESFAS instrument Functions are OPERABLE. However, these statements are redundant to the definition of a CHANNEL FUNCTIONAL TEST and ESF RESPONSE TIME provided in ITS Section 1.1 and therefore are not necessary.

The definition of a CHANNEL FUNCTIONAL TEST states, in part, that the test "…may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step." Thus, the definition, combined with the requirement to include the bypass removal function in the CHANNEL CALIBRATION, ensures the total bypass function is demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

The definition of the ESF RESPONSE TIME states, in part, that the test, "…shall be that time interval from FEATURE (ESF) RESPONSE when the monitored parameter exceeds its ESF actuation TIME setpoint at the channel sensor until the ESF equipment is capable of performing its safety function. This encompasses the requirement that at least once channel per Function be included in an ESF RESPONSE TIME. The required testing will continue, per the ITS SR wording and the definitions provided in Section 1.1, to include the features necessary to verify the ESFAS instrument Functions are OPERABLE.

This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS Table 3.3-3 for ESFAS instrumentation has three columns stating various requirements for each function. These columns are labeled, "TOTAL NO. OF CHANNELS," "CHANNELS TO TRIP," and "MINIMUM CHANNELS OPERABLE." ITS Table 3.3.3-1 does not retain the "TOTAL NO. OF CHANNELS" or "CHANNELS TO TRIP" columns. DOC LA01 describes the change that moves the information of the "TOTAL NO. OF CHANNELS" and "CHANNELS TO TRIP" columns to the Bases. This changes the CTS by changing the "MINIMUM CHANNELS OPERABLE" column to "REQUIRED CHANNELS OPERABLE" and changing the number of channels to four (4) channels for those Functions with a "MINIMUM CHANNELS OPERABLE" criteria less than four (4) channels. Additionally, the revised "MINIMUM CHANNELS OPERABLE" criteria are moved to the ITS 3.3.3. LCO statement and the "MINIMUM CHANNELS OPERABLE" column is not included in ITS Table 3.3.3-1. This change is designated as administrative because it does not result in a technical change to the CTS.

A05 Unit 1 CTS Table 3.3-3, Actions 9.b, 10A.b, 14.b, and Unit 2 CTS Table 3.3-3, Actions 13, 14.b, 20.b state, in part, that all functional units receiving an input from the inoperable channel are also bypassed or tripped. The ITS instrumentation format is presented by Functions based on parameters the channels sense and systems the instrument Functions actuate. As a result,

when a functional unit impacts OPERABILITY of multiple ESFAS Functions, each Function is addressed separately and independently depending on the nature of the inoperability. ITS 3.3.3 ACTIONS are applied to each Function, as applicable, and include a Required Action to place the affected trip unit in trip or bypass within 1 hour, as appropriate. CTS 3.0.2 (ITS LCO 3.0.2) establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. Therefore, when a common instrument becomes inoperable, LCO 3.0.2 requires each Function affected by the inoperability to be declared inoperable and the associated ACTIONS shall be met. The additional clarifying statement in the CTS Actions is not necessary and is not included in the presentation of the ITS 3.3.3 ACTIONS. This change is designated as administrative because it does not result in a technical change to the CTS.

A06 Unit 1 CTS 3.3.2.1 and Unit 2 CTS 3.3.2 do not contain a default action to perform when the Table 3.3-3 Actions cannot be completed within the required time; therefore, an entry into CTS 3.0.3 would be required. ITS 3.3.3 ACTION H requires the unit to be in MODE 3 in 6 hours and MODE 4 in 12 hours when one or more Required Actions is not met within its associated Completion Time. This changes the CTS by explicitly stating shutdown ACTIONS when actions cannot be performed within the required time.

The purpose of ITS 3.3.3 ACTION I is to ensure the plant is brought to a MODE in which the LCO for the ESFAS Functions does not apply within a reasonable amount of time in a controlled manner because the Technical Specification actions cannot be completed as required. Unit 1 CTS 3.3.2.1 and Unit 2 CTS 3.3.2 are silent on these actions, deferring to CTS 3.0.3 for the actions to accomplish this. The proposed change is acceptable because the ACTIONS specified in ITS 3.3.3 adopt the ISTS structure for placing the unit outside the MODE of Applicability without changing the time specified to enter MODE 3 or MODE 4. A default Condition eliminates the need to LCO 3.0.3 when one or more Required Actions cannot be met within the associated Completion Times. This change is designated as administrative because it does not result in a technical change to the CTS.

A07 Unit 1 CTS 3.3.2.1 Table 3.3-3 Action 14, and Unit 2 CTS 3.3.2 Table 3.3-3 Action 20 provide actions when one channel of an AFAS Function is inoperable. CTS 3.3.2 Table 3.3-3 Action 13 also provides actions when one channel of an SIAS, CIAS, or MSIS Function is inoperable. These CTS Actions state in part, “The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.” ITS 3.3.3 ACTIONS C and D include this action to restore the channel to OPERABLE with a Completion Time of, “Prior to entering MODE 3 following the following next MODE 5 entry.” This presentation preference is similar to the Completion Time specified in ISTS 3.3.5 (Digital), Required Action A.2. ITS 3.3.3, Conditions C and D are modified by a Note which requires the action to restore the channel to OPERABLE status within the Completion Time to be performed whenever the Condition is entered. This changes the CTS by explicitly stating that the Required Action must be performed within the Completion Time whenever the condition is entered.

This change is necessary to eliminate the conflict with the general CTS 3.0.2 (ITS LCO 3.0.2) requirement that if the LCO is met or is no longer applicable prior to expiration of the specified time interval(s), completion of the ACTIONS is not required, unless otherwise stated. The addition of the Note constitutes an "unless otherwise stated." The Note will require the action to be completed within the Completion Time even when the LCO is no longer applicable, to ensure the channel is restored to OPERABLE prior to the next reactor startup from MODE 5 conditions. This change is acceptable because it reflects the current understanding and application of the CTS Actions requiring the applicable channel to be restored to OPERABLE status whenever the LCO is not met.

This change is designated as administrative as it clarifies the current understanding of the CTS requirement while providing an "unless otherwise stated," to the requirements of CTS 3.0.2 (ITS LCO 3.0.2).

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.3.2.1 Action a. and Unit 2 CTS 3.3.2 Action a. state, in part, that with an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable. ITS Table 3.3.3-1 Footnote (a) states, "If the as-found channel setpoint is outside its predefined as-found acceptance criteria, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service." ITS 3.3.3 Bases, referring to the Allowable Values, states: "This value needs to be specified in the Technical Specifications in order to define OPERABILITY of the channels and is designated as the Allowable Value." ITS Section 3.3.3 Bases also states, "If the actual setting (as-found setpoint) of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found acceptance criteria band (e.g., conservative side of the as-found acceptance criteria band), the channel is OPERABLE, but degraded. The degraded condition will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the Nominal Trip Setpoint (NTSP) (within the allowed tolerance) and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance." This changes the CTS by requiring an evaluation of channel functionality (extent of which is expanded on in the TS Bases) prior to returning it to service in addition to the as-found value being conservative to the Allowable Value.

The purpose of this Note is to address a concern that the Technical Specification requirements for Limiting Safety System Settings (LSSS) may not be fully in compliance with the intent of 10 CFR 50.36. Specifically, the concern is that the existing Surveillance Requirements may not provide adequate assurance that instruments will actuate safety functions at the point assumed in the applicable safety analysis. 10 CFR 50.36(c)(1)(ii)(A) states, "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the settings must be so chosen that automatic protective action will correct the

abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor." The proposed change clarifies the Technical Specification requirements to ensure that the automatic protective action will correct the abnormal situation before a safety limit is exceeded. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. This change is consistent with the ISTS. This change is considered a more restrictive change because additional requirements have been added to Surveillance Requirements.

M02 Unit 1 CTS 3.3.2.1 Action a. and Unit 2 CTS 3.3.2 Action a. state, in part, that with an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status "with its trip setpoint adjusted consistent with the Trip Setpoint value." ITS Table 3.3.3-1 Footnote (b) states, "The instrument channel setpoint shall be reset to a value that is within the as-left acceptance criteria around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall not be returned to OPERABLE status. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left acceptance criteria apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP and the methodologies used to determine the as-found and as-left acceptance criteria are specified in the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report." This changes the CTS by providing more detailed information describing what "consistent with the Trip Setpoint" means and states a specific location where the methodology for determining the double-sided NTSP instrument tolerance bands is located.

The purpose of this Note is to address a concern that the Technical Specification requirements for LSSS may not be fully in compliance with the intent of 10 CFR 50.36. Specifically, the concern is that the existing Surveillance Requirements may not provide adequate assurance that instruments will always actuate safety functions at the point assumed in the applicable safety analysis. 10 CFR 50.36(c)(1)(ii)(A) states, "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the settings must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor." The proposed change

clarifies the Technical Specification requirements to ensure that the automatic protective action will correct the abnormal situation before a safety limit is exceeded. This change is consistent with the ISTS and is considered a more restrictive change because additional requirements have been added to Surveillance Requirements.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS 3.3.2.1 and Unit 2 CTS 3.2.2 Table 3.3-4, ESFAS Instrumentation Trip Values, has two columns stating trip setpoints and allowable values for each Function. Unit 1 CTS 3.3.2.1 and Unit 2 CTS 3.2.2 require the ESFAS instrumentation setpoints to be set consistent with the Trip Setpoint values shown in Table 3.3-4. However, the CTS Action is only required to be taken when the setpoint is less conservative than the Allowable Value column of Table 3.3-4. When the setpoint is less conservative than the Allowable Value, the channel is to be declared inoperable and adjusted consistent with the Trip Setpoint value. CTS Table 3.3-4 specifies both the Trip Setpoints and Allowable Values for the ESFAS Instrumentation Functional Units. ITS 3.3.3 requires the ESFAS instrumentation for each Function in Table 3.3.3-1 to be OPERABLE. ITS Table 3.3.3-1 specifies only the Allowable Values for the ESFAS Instrumentation Functions. The Allowable Values represent the OPERABILITY limit of the channels in ITS. This changes the CTS by moving the Trip Setpoints to the Technical Requirements Manual (TRM). R1 R1

The purpose of the trip setpoint requirements is to ensure required automatic safety systems are actuated to protect against violating core design limits, breaching the Reactor Coolant System pressure boundary, and to mitigate accidents. Pursuant to 10 CFR 50.36(c)(1)(ii)(A), if it is determined that an automatic protective device for a variable on which a safety limit has been placed (i.e., limiting safety system setting) does not function as required, appropriate action is taken to ensure the abnormal situation is corrected before a safety limit is exceeded, which may include shutting down the reactor. The PSL Instrument Setpoint Methodology calculates Nominal Trip Setpoints (NTSPs) using methods consistent with the guidance provided in NRC Regulatory Guide 1.105, “Setpoints For Safety-Related Instrumentation,” and ANSI/ISA Standard 67.04, "Setpoints for Nuclear Safety-Related Instrumentation." Additionally, pre-defined limits (double-sided Operability limits and as-left limits) are determined for each instrument consistent with the guidance provided in NRC Regulatory Guide 1.105 and ANSI/ISA–RP67.04, “Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation.” The instrument Operability limit band in plant uncertainty calculations is synonymous with the as-found acceptance criteria band specified in ITS and is centered about the nominal equipment setting (clarified in calculations as the field trip setpoint or NTSP). The PSL

Instrument Setpoint Methodology, including the method of determining instrument uncertainties, was reviewed by the NRC during the review of Extended Power Uprate (EPU) as documented in the EPU safety evaluations for License Amendments 213 (Unit 1) and 163 (Unit 2), dated July 9, 2012 and September 24, 2012, respectively (NRC ADAMS Accession Nos. ML12181A019 and ML12235A463). The NRC staff determined that the PSL Instrument Setpoint Methodology described in the EPU license request meets the requirements of 10 CFR 50, Appendix A, General Design Criterion 13 and the regulatory guidance in RG 1.105 (EPU SE pg. 92 (Unit 1) and 95 (Unit 2)). The NRC staff also determined that the application of the clarifications of RG 1.105 in Regulatory Issue Summary 2006-17 "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," is acceptable (EPU SE pg. 94 (Unit 1) and 97 (Unit 2)). | R1

The removal of these details for meeting Technical Specification requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the Allowable Values associated with the ESFAS Instrumentation, which are designated as the Operability limits for the required instrument Functions. Footnotes (a) and (b) in Table 3.3.3-1 are added ensure channel performance continues to verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology consistent with the NRC guidance specified in Regulatory Issue Summary 2006-17. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 (*Type 1 – Removing Details of System Design and System Description, Including Design Limits*) CTS Table 3.3-3 for ESFAS instrumentation has three columns stating various requirements for each function. These columns are labeled, "TOTAL NO. OF CHANNELS," "CHANNELS TO TRIP," and "MINIMUM CHANNELS OPERABLE." ITS Table 3.3.3-1 does not retain the "TOTAL NO. OF CHANNELS" or "CHANNELS TO TRIP" columns. This changes the CTS by moving the information of the "TOTAL NO. OF CHANNELS" and "CHANNELS TO TRIP" columns to the Bases. The MINIMUM CHANNELS OPERABLE" column is deleted as described in discussion of change (DOC) A04. | R1

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the number of required channels and the appropriate Condition to enter if a required channel becomes inoperable. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program

in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA03 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 3.3-3 Action 14.a (Unit 1) and 20.a (Unit 2) require, in part, AFAS-1 or AFAS-2 associated with the inoperable channel to be placed in the bypass position when a steam generator (SG) level AFAS channel cannot be restored to OPERABLE status within the 48 hours. These CTS actions also require both AFAS-1 and AFAS-2 associated with the inoperable channel to be placed in the bypass position when a SG differential pressure or feedwater header differential pressure AFAS channel cannot be restored to OPERABLE status within the 48 hours. ITS 3.3.3 Required Action B.2.2 requires, in part, to place the affected trip unit in bypass as an option to restoring an AFAS instrument channel to OPERABLE status within 48 hours. ITS 3.3.3 ACTION B does not include the procedural details on which trip units to place in bypass (i.e., AFAS-1 or AFAS-2) based on the inoperable AFAS function. This changes the CTS by moving the procedural detail on what AFAS channels to bypass to the Bases.

The removal of these details, which are related to details on how to perform a Technical Specification Required Action, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to bypass the affected trip unit when a AFAS Function instrument channel is inoperable. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to how to perform an action is being removed from the Technical Specifications.

LA04 **Unit 2 only:** (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements*) CTS Table 3.3-3, Actions 13, 14.b, and 20.b state, in part, that all functional units receiving an input from the inoperable channel are also bypassed or tripped. CTS Table 3.3-3, Actions 13 and 14.b provide a process instrument and Function bypass list that identifies the associated Functions that must be bypassed if the instrument becomes inoperable, and Action 20.b references the Action 13 list. ITS 3.3.3 ACTIONS for each Function include a Required Action to place the affected trip unit in trip or bypass within 1 hour. CTS LCO 3.0.2 and ITS LCO 3.0.2 establish that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. Therefore, when an instrument channel becomes inoperable, LCO 3.0.2 requires each Function affected by the inoperability shall be declared inoperable and the associated ACTIONS shall be met. The CTS process instrument and Function bypass list is not necessary in the Specifications. This information is removed from the Technical Specifications and moved to the UFSAR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that all functional units receiving an input from an inoperable channel are bypassed or tripped. ITS 3.3.3 ACTIONS for each Function include a Required Action to place the affected trip unit in trip or bypass within 1 hour. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* Unit 1 CTS 3.3.2.1 and Unit 2 CTS 3.3.2 require the MSIS Steam Generator Pressure – Low Function (Table 3.3-3 Functional Unit 4.b) to be OPERABLE in MODES 1, 2, and 3 except when bypassed as noted by Note (c). Unit 2 CTS 3.3.2 also requires the MSIS Containment Pressure – High channel to be OPERABLE in MODES 1, 2, and 3. ITS 3.3.3 also requires the same MSIS Function to be OPERABLE. However, Table 3.3.3-1, Function 4.a (MSIS Steam Generator Pressure – Low) modifies MODES 2 and 3 by Footnote (f) that states that the Main Steam Isolation Signal (MSIS) Function Steam Generator Pressure - Low signal is not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed. Unit 2 Table 3.3.3-1, Functions 4.a (MSIS Steam Generator Pressure – Low) and 4.b (MSIS Containment Pressure – High) modifies MODES 2 and 3 by Footnote (f) that states that the Main Steam Isolation Signal (MSIS) Function Steam Generator Pressure - Low and Containment Pressure - High signals are not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed. This changes the CTS by relaxing the MODE 2 and 3 requirements of the MSIS Steam Generator Pressure – Low instrument Function and the Unit 2 MSIS Containment Pressure - High instrument Function.

The purpose of the Applicability of the MSIS Steam Generator Pressure – Low and Unit 2 MSIS Containment Pressure - High instrument Functions is to ensure the ESFAS MSIS Functions are OPERABLE when the supported equipment (i.e., main steam isolation valves (MSIVs) and main feedwater isolation valves (MFIVs)) is required to be OPERABLE. Automatic steam line and feedwater line isolation is assumed in the mitigation of a major secondary system pipe rupture accident (e.g., main steam line break or feedwater line break). In MODES 1, 2, and 3, there is sufficient energy in the primary and secondary systems to warrant automatic ESF System response to ensure the main steam and feedwater lines can be isolated in the event of a high energy secondary system pipe rupture. The exception to MODE 2 and 3 is added to clarify that the MSIS Functions are not required to be OPERABLE when the valves actuated by the MSIS instrumentation are in a position that supports the safety analyses. When the valves isolated by the MSIS Functions are in the closed position, they are in their assumed accident position. This change is acceptable, because when the

MSIVs and MFIVs are closed, the adverse effects of a high energy secondary system pipe rupture are precluded and the requirements continue to ensure that the structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses. The proposed applicability change provides the same level of protection as the current requirements for the supported equipment and is consistent with the applicability for Function 4.a in ISTS Table 3.3.4-1 (Analog) and Function 4.b in ISTS Table 3.3.5-1 (Digital). This change is designated as less restrictive because the ITS LCO requirements are applicable in fewer operating conditions than in the CTS.

L02 (*Category 4 – Relaxation of Required Action*) Unit 1 CTS Table 3.3-3 Actions 9.c and 13, which apply to the SIAS, CIS and MSIS Functions and the RAS Function, respectively, require the minimum number of channels to be OPERABLE (i.e., three of four channels) except an additional channel may be bypassed for up to 48 hours (Action 9) or 2 hours (Action 13) while performing tests and maintenance on that channel provided the other inoperable channel is placed in the trip condition. Unit 2 CTS Table 3.3-3 Action 19, which applies to the RAS Function, requires the minimum number of channels to be OPERABLE (i.e., three of four channels) except an additional channel may be bypassed for up to 2 hours while performing testing per Specification 4.3.2.1. ITS 3.3.3 ACTION F provides the same condition (i.e., two of four channels inoperable) for the SIAS, CIS (Unit 1), CIAS (Unit 2), MSIS, and RAS Functions. When one or more Functions (except CSAS or AFAS) with two ESFAS trip units or associated instrument channels are inoperable, one trip unit must be placed in bypass and the other trip unit must be placed in trip within 1 hour and one trip unit must be restored to OPERABLE status within 48 hours. This changes the CTS by allowing two ESFAS Function trip units or associated instrument channels to be inoperable for reasons other than maintenance and testing and extends the time to restore the Unit 2 RAS ESFAS Function to OPERABLE status from 2 hours to 48 hours.

The purpose of the CTS Actions is to allow an additional ESFAS channel to be inoperable for the purposes of testing. The SIAS, CIS (Unit 1), CIAS (Unit 2), MSIS, and RAS ESFAS Function channels are configured in a two-out of-four coincidence. The two-out-of-four ESFAS logic is changed to a two-out-of-three logic for a given input parameter in one channel at a time by bypassing one channel input to the logic. With one channel of protective instrumentation bypassed, the ESFAS Function is in two-out-of-three logic, but with another channel failed the ESFAS may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESFAS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, ESFAS actuation will occur. Therefore, it is also acceptable to allow an additional channel to be inoperable for reasons other than testing provided one channel is in bypass and the other channel is in trip. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the remaining ESFAS

instrument channels, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. This change also provides appropriate remedial actions in lieu of requiring initiation of an unnecessary plant transient per ITS LCO 3.0.3 (CTS 3.0.3). The 48-hour Completion Time is based upon operating experience, which has demonstrated that a random failure of another channel occurring during the 48 hour period is a low probability event. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 (*Category 4 – Relaxation of Required Action*) Unit 1 CTS 3.3.2.1 and Unit 2 CTS 3.3.2 do not explicitly provide requirements associated with the bypass removal feature associated with the SIAS or MSIS ESFAS Functions. However, CTS 4.3.2.1.2 and Unit 2 CTS 4.3.2.2 require channel functional and calibration testing of the channel bypass logic, which includes the bypass removal feature. Therefore, per the requirements of CTS 4.0.1 (ITS SR 3.0.1), CTS 3.3.2.1 (Unit 1) and CTS 3.3.2 (Unit 2) encompass the ESFAS channel bypass removal feature. If an ESFAS bypass removal channel is inoperable, CTS 3.0.3 is entered since there is no Condition and Action specified in the CTS. ITS 3.3.3 also includes Surveillances for the channel bypass removal feature. In addition, ITS 3.3.3 ACTIONS G and H are added to provide remedial measures when the bypass removal capability is inoperable to an SIAS or MSIS ESFAS Function channel. ITS Required Actions G.1 and H.1 require disabling the inoperable bypass channel within 1 hour when one or two bypass removal channels is inoperable. When one bypass removal channel is inoperable, ITS 3.3.3 ACTION G alternately allows placing the affected channel in bypass or trip within 1 hour and restoring the channel to OPERABLE status within 48 hours. When two bypass removal channels are inoperable, ITS 3.3.3 ACTION H alternately allows placing one channel in trip and the other channel in bypass and restoring one channel to OPERABLE status within 48 hours. This changes the CTS by adding specific Technical Specification actions when one or two bypass removal channels are inoperable to the SIAS or MSIS ESFAS Functions.

The purpose of the proposed actions is to provide appropriate remedial actions in the event the automatic bypass removal feature is inoperable. With one or two bypass removal channels inoperable, it is permissible to continue operation with the bypass permissive removal channels failed, providing the bypass is disabled. This can be accomplished by removing the bypass with the manual bypass key switch, which disables the bypass in its respective train. Since the bypass Function must be manually enabled, the bypass permissive Function will not by itself cause an undesired bypass insertion. Alternatively, the bypass may be disabled by defeating the bypass permissive input in one or two of the four channels to the two-out-of-four bypass removal logic, placing the bypass removal feature in one-out-of-three logic. Thus, any of the remaining channels is capable of removing the bypass feature when the bypass enable conditions are no longer valid. The Completion Times are consistent with the completion times for one or two inoperable SIAS or MSIS ESFAS Function channels. This change is also consistent with the ISTS.

This change is considered acceptable because the proposed remedial actions will ensure the related ESFAS Functions can continue to perform their related safety function when required by the plant design basis and safety analysis by requiring the bypass removal channel to be disabled or take action consistent with the current actions for an inoperable ESFAS channel. This change also provides appropriate remedial actions in lieu of requiring initiation of an unnecessary plant transient per ITS LCO 3.0.3 (CTS 3.0.3). This change is designated as less restrictive because the ITS ACTIONS provide remedial actions that are less restrictive than the CTS Action requiring a plant shutdown.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~4~~ Engineered Safety Features Actuation System (ESFAS) Instrumentation ~~(Analog)~~

3.3.2.1 LCO 3.3.~~4~~ Four ESFAS trip units and associated instrument and bypass removal channels for each Function in Table 3.3.~~4~~-1 shall be OPERABLE.

Table 3.3-3 Functional Unit 1.b, 1.c, 2.b, 3.b, 3.c, 4.a, 5.b, 7.c, 8.a, 8.b

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------

DOC A02 Separate Condition entry is allowed for each ESFAS Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-3 Action 10A A. ~~[~~One Containment Spray Actuation Signal (CSAS) trip unit or associated instrument inoperable. INSERT 1	A.1 Place affected trip unit in bypass. or trip AND A.2.1 Restore channel to OPERABLE status. 48 hours OR A.2.2 Place affected trip unit in trip. 48 hours	1 hour ~~]~~
Table 3.3-3 Action 9 ~~B~~. D One or more Functions with one ESFAS trip unit or associated instrument channel (except CSAS) inoperable. , RAS or AFAS	~~B~~.1 D Place affected trip unit in bypass or trip. AND ~~B~~.2.1 D Restore channel to OPERABLE status. OR	1 hour ~~[~~48~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~

1

INSERT 1

Table 3.3-3 Action 13	B. One Containment Sump Recirculation Actuation Signal (RAS) trip unit or associated instrument inoperable.	B.1	Place affected trip unit in bypass or trip.	1 hour
		AND		
		B.2	Restore trip unit to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program
Table 3.3-3 Action 14.a, 14.b	C. ----------NOTE---------- Required Action C.2.2.2 must be completed whenever this Condition is entered. ---------- One or more Functions with one Auxiliary Feedwater Actuation Signal (AFAS) trip unit or associated instrument inoperable.	C.1	Place affected trip unit in bypass or trip.	1 hour
		AND		
		C.2.1	Restore channel to OPERABLE status.	48 hours
		OR		
		C.2.2.1	Place affected trip unit in bypass.	48 hours
		AND		
		C.2.2.2	Restore channel to OPERABLE status.	Prior to entering MODE 3 following next MODE 5 entry

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~B~~.2.2 ~~[~~ Place affected trip unit in trip.	48 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
~~C~~. One or more Functions with two ESFAS trip units or associated instrument channels (except CSAS) inoperable.	~~C~~.1 Place one trip unit in bypass and place the other trip unit in trip. AND ~~C~~.2 Restore one trip unit to OPERABLE status.	1 hour ~~[48]~~ hours
~~D~~. One or more Functions with one automatic bypass removal channel inoperable.	~~D~~.1 Disable bypass channel. OR ~~D~~.2.1 Place affected trip units in bypass or trip. AND ~~D~~.2.2.1 Restore bypass removal channel and affected trip units to OPERABLE status. OR	1 hour 1 hour ~~[48]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~

1

INSERT 2

Table 3.3-3 Action 10B Action 15	E. One or more Functions with two or more AFAS trip units or associated instrument channels inoperable. OR Two CSAS trip units or associated instrument channels inoperable.	E.1 Place one trip unit in bypass and place the other trip unit in trip. AND E.2 Restore one trip unit to OPERABLE status.	1 hour 48 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
SIAS or MSIS	~~D~~.2.2.2 ~~[~~ Place affected trip units in trip.	48 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
DOC M01 ~~E~~. One or more Functions with two automatic bypass removal channels inoperable.	~~E~~.1 Disable bypass channels. OR ~~E~~.2.1 Place one affected trip unit in bypass and place the other in trip for each affected ESFAS Function. AND ~~E~~.2.2 ~~[~~ Restore one bypass channel and the associated trip unit to OPERABLE status for each affected trip Function.	1 hour 1 hour 48 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
Action 10B, 13, 15 ~~F~~. Required Action and associated Completion Time not met.	~~F~~.1 Be in MODE 3. AND ~~F~~.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.3.2.1.1	SR 3.3.~~4~~.1	Perform a CHANNEL CHECK of each ESFAS instrument channel.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
4.3.2.1.1	SR 3.3.~~4~~.2	Perform a CHANNEL FUNCTIONAL TEST of each ESFAS instrument channel.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
4.3.2.1.2	SR 3.3.~~4~~.3	Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function.	~~Once within 92 days prior to each reactor startup~~ In accordance with the Surveillance Frequency Control Program
4.3.2.1.1	SR 3.3.~~4~~.4	Perform a CHANNEL CALIBRATION of each ESFAS instrument channel, including bypass removal functions.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.~~4~~.5	Verify ESF RESPONSE TIME is within limits.	~~[[18] months on a STAGGERED TEST BASIS~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

Table 3.3.~~4~~-1 (page 1 of 2)
Engineered Safety Features Actuation System Instrumentation

	FUNCTION	MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
	1. Safety Injection Actuation Signal (SIAS)			
Table 3.3-3 Function 1.b	a. Containment Pressure - High	1,2,3	SR 3.3.~~4~~.1 SR 3.3.~~4~~.2(a)(b) SR 3.3.~~4~~.4(a)(b) SR 3.3.~~4~~.5	≤ ~~[19.0] psia~~ 5 psig
Table 3.3-3 Function 1.c	b. Pressurizer Pressure - Low(c)	1,2,3	SR 3.3.~~4~~.1 SR 3.3.~~4~~.2(a)(b) SR 3.3.~~4~~.3 SR 3.3.~~4~~.4(a)(b) SR 3.3.~~4~~.5	≥ ~~[1687]~~ psia 1600
	2. Containment Spray Actuation Signal(d)			
Table 3.3-3 Function 2.b	a. Containment Pressure - ~~High~~ High-High	1,2,3	SR 3.3.~~4~~.1 SR 3.3.~~4~~.2(a)(b) SR 3.3.~~4~~.4(a)(b) SR 3.3.~~4~~.5	≤ ~~[19.0] psia~~ 10 psig
	3. Containment Isolation ~~Actuation~~ Signal			
Table 3.3-3 Function 3.b	a. Containment Pressure - High	1,2,3	SR 3.3.~~4~~.1 SR 3.3.~~4~~.2(a)(b) SR 3.3.~~4~~.4(a)(b) SR 3.3.~~4~~.5	≤ ~~[19.0]~~ psia 5 psig
Table 3.3-3 Function 3.c	~~[~~ b. Containment Radiation - High	1,2,3	SR 3.3.~~4~~.1 SR 3.3.~~4~~.2(a)(b) SR 3.3.~~4~~.4(a)(b) SR 3.3.~~4~~.5	≤ ~~[2x Background]]~~ 10 R/hr

DOC M01 (a) If the as-found channel setpoint is outside its predefined ~~as-found tolerance~~ acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

DOC M02 (b) The instrument channel setpoint shall be reset to a value that is within the as-left ~~tolerance~~ acceptance criteria band around the ~~Limiting~~ Nominal Trip Setpoint (~~LTSP~~ NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the ~~LTSP~~ NTSP are acceptable provided that the as-found and as-left ~~tolerances~~ acceptance criteria bands apply to the actual setpoint implemented in the Surveillance procedures (~~Nominal Trip Setpoint~~ NTSP) to confirm channel performance. The ~~LTSP~~ NTSP and the methodologies used to determine the as-found and as-left ~~tolerances~~ acceptance criteria bands are specified in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~ the Technical Requirements Manual and Section 7.3 of the Updated Final Safety Analysis Report, respectively.

Table 3.3-3 Note (a) (c) Pressurizer Pressure - Low may be manually bypassed when pressurizer pressure is < ~~[1800]~~ 1725 psia. The bypass shall be automatically removed whenever pressurizer pressure is ≥ ~~[1800]~~ 1725 psia.

Table 3.3-3 Note (b) ~~[~~ (d) SIAS is also required as a permissive to initiate containment spray. ~~]~~

Table 3.3.4-1 (page 2 of 2)
Engineered Safety Features Actuation System Instrumentation

FUNCTION	MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Main Steam Isolation Signal			585 psig
a. Steam Generator Pressure - Low[(e)] (Table 3.3-3 Function 4.b)	1,2[(f)],3[(f)]	SR 3.3.4.1 SR 3.3.4.2[(a)(b)] SR 3.3.4.3 SR 3.3.4.4[(a)(b)] SR 3.3.4.5	≥ ~~[495] psig~~
5. Recirculation Actuation Signal			48
a. Refueling Water Tank Level - Low (Table 3.3-3 Function 5.b)	1,2,3	[SR 3.3.4.1] SR 3.3.4.2[(a)(b)] SR 3.3.4.4[(a)(b)] SR 3.3.4.5	~~[~~≥ ~~24~~ inches ~~and ≤ 30] inches~~ above tank bottom
6. Auxiliary Feedwater Actuation Signal (AFAS)			
a. Steam Generator A Level - Low (Table 3.3-3 Function 7.c)	1,2,3	SR 3.3.4.1 SR 3.3.4.2[(a)(b)] SR 3.3.4.4[(a)(b)] SR 3.3.4.5	≥ ~~[45.7]~~ % (18)
b. Steam Generator B Level - Low (Table 3.3-3 Function 7.c)	1,2,3	SR 3.3.4.1 SR 3.3.4.2[(a)(b)] SR 3.3.4.4[(a)(b)] SR 3.3.4.5	≥ ~~[45.7]~~ % (18)
c. Steam Generator Pressure Difference - High (A > B) or (B > A) (Table 3.3-3 Function 8.a)	1,2,3	SR 3.3.4.1 SR 3.3.4.2[(a)(b)] SR 3.3.4.4[(a)(b)] SR 3.3.4.5	~~≤ [48.3]~~ psid ≥ 89.2 and ≤ 281
d. Feedwater Header Pressure Difference – High (Table 3.3-3 Function 8.b)	1, 2, 3	SR 3.3.3.1, SR 3.3.3.2 SR3.3.3.4, SR 3.3.3.5	≥ 56.0 and ≤ 157.5 psid

DOC M01 (a) If the as-found channel setpoint is outside its predefined ~~as-found tolerance~~, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

DOC M02 (b) The instrument channel setpoint shall be reset to a value that is within the as-left ~~tolerance~~ around the ~~Limiting~~ Trip Setpoint (~~LTSP~~) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the ~~LTSP~~ are acceptable provided that the as-found and as-left ~~tolerances~~ apply to the actual setpoint implemented in the Surveillance procedures ~~(Nominal Trip Setpoint)~~ to confirm channel performance. The ~~LTSP~~ and the methodologies used to determine the as-found and as-left ~~tolerances~~ are specified in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~.

Table 3.3-3 Note (c) (e) Steam Generator Pressure - Low may be manually bypassed when steam generator pressure is < ~~[785] psia~~. The bypass shall be automatically removed whenever steam generator pressure is ≥ ~~[785] psia~~.

DOC L02 (f) ~~Only the~~ Main Steam Isolation Signal (MSIS) Function ~~and the~~ Steam Generator Pressure - Low ~~and Containment Pressure - High~~ signal~~s~~ ~~are~~ not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed ~~and [de-activated]~~.

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~4~~ Engineered Safety Features Actuation System (ESFAS) Instrumentation ~~(Analog)~~

3.3.2 LCO 3.3.~~4~~ Four ESFAS trip units and associated instrument and bypass removal channels for each Function in Table 3.3.~~4~~-1 shall be OPERABLE.

Table 3.3-3 Functional Unit 1.b, 1.c, 2.b, 3.b, 3.c, 3.d, 4.b, 4.c, 5.b, 7.c, 8.a, 8.b

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------

DOC A02 Separate Condition entry is allowed for each ESFAS Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-3 Action 18A A. ~~[~~ One Containment Spray Actuation Signal (CSAS) trip unit or associated instrument inoperable. INSERT 1	A.1 Place affected trip unit in bypass: or trip AND A.2.1 Restore channel to OPERABLE status. 48 hours OR A.2.2 Place affected trip unit in trip. 48 hours	1 hour ~~]~~
Table 3.3-3 Action 13 ~~B~~. D One or more Functions with one ESFAS trip unit or associated instrument channel (except CSAS) inoperable. , RAS or AFAS ----------NOTE---------- Required Action D.2 must be completed whenever this Condition is entered. ----------	~~B~~.1 D Place affected trip unit in bypass or trip. AND ~~B~~.2.~~1~~ D Restore channel to OPERABLE status. ~~OR~~	1 hour ~~[48] hours~~ ~~[OR~~ Prior to entering MODE 3 following next MODE 5 entry ~~In accordance with the Risk Informed Completion Time Program]~~

1

INSERT 1

Table 3.3-3 Action 19	B. One Containment Sump Recirculation Actuation Signal (RAS) trip unit or associated instrument inoperable.	B.1 Place affected trip unit in bypass or trip. AND B.2 Restore trip unit to OPERABLE status.	1 hour 48 hours OR In accordance with the Risk Informed Completion Time Program
Table 3.3-3 Action 20.a 20.b	C. ----------NOTE---------- Required Action C.2.2.2 must be completed whenever this Condition is entered. ---------- One or more Functions with one Auxiliary Feedwater Actuation Signal (AFAS) trip unit or associated instrument inoperable.	C.1 Place affected trip unit in bypass or trip. AND C.2.1 Restore channel to OPERABLE status. OR C.2.2.1 Place affected trip unit in bypass. AND C.2.2.2 Restore channel to OPERABLE status.	1 hour 48 hours 48 hours Prior to entering MODE 3 following next MODE 5 entry

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~B.2.2~~ ~~[Place affected trip unit in trip.~~	~~48 hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
~~C~~. One or more Functions with two ESFAS trip units or associated instrument channels (except CSAS) inoperable.	~~C~~.1 Place one trip unit in bypass and place the other trip unit in trip. AND ~~C~~.2 Restore one trip unit to OPERABLE status.	1 hour ~~[~~48~~]~~ hours
~~D~~. One or more Functions with one automatic bypass removal channel inoperable.	~~D~~.1 Disable bypass channel. OR ~~D~~.2.1 Place affected trip units in bypass or trip. AND ~~D~~.2.2.1 Restore bypass removal channel and affected trip units to OPERABLE status. OR	1 hour 1 hour ~~[~~48~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~

1

INSERT 2

Table 3.3-3 Action 18B Action 21	E. One or more Functions with two or more AFAS trip units or associated instrument channels inoperable. OR Two CSAS trip units or associated instrument channels inoperable.	E.1 Place one trip unit in bypass and place the other trip unit in trip. AND E.2 Restore one trip unit to OPERABLE status.	1 hour 48 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~D~~.2.2.2 ~~[~~ Place affected trip units in trip.	48 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
SIAS or MSIS ~~E~~. One or more Functions with two automatic bypass removal channels inoperable.	~~E~~.1 Disable bypass channels. OR ~~E~~.2.1 Place one affected trip unit in bypass and place the other in trip for each affected ESFAS Function. AND ~~E~~.2.2 ~~[~~ Restore one bypass channel and the associated trip unit to OPERABLE status for each affected trip Function.	1 hour 1 hour 48 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
~~F~~. Required Action and associated Completion Time not met.	~~F~~.1 Be in MODE 3. AND ~~F~~.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.3.2.1	SR 3.3.~~4~~.1	Perform a CHANNEL CHECK of each ESFAS instrument channel.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
4.3.2.1	SR 3.3.~~4~~.2	Perform a CHANNEL FUNCTIONAL TEST of each ESFAS instrument channel.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
4.3.2.2	SR 3.3.~~4~~.3	Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function.	~~Once within 92 days prior to each reactor startup~~ In accordance with the Surveillance Frequency Control Program
4.3.2.1	SR 3.3.~~4~~.4	Perform a CHANNEL CALIBRATION of each ESFAS instrument channel, including bypass removal functions.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.~~4~~.5	Verify ESF RESPONSE TIME is within limits.	~~[[18] months on a STAGGERED TEST BASIS~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

Table 3.3.~~4~~-1 (page 1 of 2)
Engineered Safety Features Actuation System Instrumentation

	FUNCTION	MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
	1. Safety Injection Actuation Signal (SIAS)			
Table 3.3-3 Function 1.b	a. Containment Pressure - High	1,2,3	SR 3.3.~~4~~.1 SR 3.3.~~4~~.2(a)(b) SR 3.3.~~4~~.4(a)(b) SR 3.3.~~4~~.5	≤ ~~[19.0] psia~~ 3.6 psig
Table 3.3-3 Function 1.c	b. Pressurizer Pressure - Low(c)	1,2,3	SR 3.3.~~4~~.1 SR 3.3.~~4~~.2(a)(b) SR 3.3.~~4~~.3 SR 3.3.~~4~~.4(a)(b) SR 3.3.~~4~~.5	≥ ~~[1687]~~ psia 1728
	2. Containment Spray Actuation Signal(d)			
Table 3.3-3 Function 2.b	a. Containment Pressure - ~~High~~ High-High	1,2,3	SR 3.3.~~4~~.1 SR 3.3.~~4~~.2(a)(b) SR 3.3.~~4~~.4(a)(b) SR 3.3.~~4~~.5	≤ ~~[19.0] psia~~ 5.5 psig
	3. Containment Isolation Actuation Signal			
Table 3.3-3 Function 3.c	a. Containment Pressure - High	1,2,3	SR 3.3.~~4~~.1 SR 3.3.~~4~~.2(a)(b) SR 3.3.~~4~~.4(a)(b) SR 3.3.~~4~~.5	≤ ~~[19.0]~~ psia 3.6 psig
Table 3.3-3 Function 3.d	~~[~~ b. Containment Radiation - High	1,2,3	SR 3.3.~~4~~.1 SR 3.3.~~4~~.2(a)(b) SR 3.3.~~4~~.4(a)(b) SR 3.3.~~4~~.5	≤ ~~[2x Background]]~~ 10 R/hr

DOC M01 (a) If the as-found channel setpoint is outside its predefined ~~as-found tolerance~~ acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

DOC M02 (b) The instrument channel setpoint shall be reset to a value that is within the as-left ~~tolerance~~ acceptance criteria band around the ~~Limiting~~ Nominal Trip Setpoint (~~LTSP~~ NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the ~~LTSP~~ NTSP are acceptable provided that the as-found and as-left ~~tolerances~~ acceptance criteria bands apply to the actual setpoint implemented in the Surveillance procedures (~~Nominal Trip Setpoint~~ NTSP) to confirm channel performance. The ~~LTSP~~ NTSP and the methodologies used to determine the as-found and as-left ~~tolerances~~ acceptance criteria bands are specified in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~ the Technical Requirements Manual and Section 7.3 of the Updated Final Safety Analysis Report, respectively.

Table 3.3-3 Note (a) (c) Pressurizer Pressure - Low may be manually bypassed when pressurizer pressure is < ~~[1800]~~ 1836 psia. The bypass shall be automatically removed whenever pressurizer pressure is ≥ ~~[1800]~~ 1836 psia.

Table 3.3-3 Note (b) ~~[~~ (d) SIAS is also required as a permissive to initiate containment spray. ~~]~~

Table 3.3.4-1 (page 2 of 2)
Engineered Safety Features Actuation System Instrumentation

FUNCTION	MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Main Steam Isolation Signal			
a. Steam Generator Pressure - Low[e]	1,2[f],3[f]	SR 3.3.4.1 SR 3.3.4.2[a][b] SR 3.3.4.3 SR 3.3.4.4[a][b] SR 3.3.4.5	≥ ~~[495] psig~~
b. Containment Pressure - High	1,2[f],3[f]	SR 3.3.3.1, SR 3.3.3.2, SR 3.3.3.4, SR 3.3.3.5	≤ 3.6 psig
5. Recirculation Actuation Signal			
a. Refueling Water Tank Level - Low	1,2,3	[SR 3.3.4.1] SR 3.3.4.2[a][b] SR 3.3.4.4[a][b] SR 3.3.4.5	~~[≥ 24 inches and ≤ 30] inches~~ above tank bottom
6. Auxiliary Feedwater Actuation Signal (AFAS)			
a. Steam Generator A Level - Low	1,2,3	SR 3.3.4.1 SR 3.3.4.2[a][b] SR 3.3.4.4[a][b] SR 3.3.4.5	≥ ~~[45.7]~~ %
b. Steam Generator B Level - Low	1,2,3	SR 3.3.4.1 SR 3.3.4.2[a][b] SR 3.3.4.4[a][b] SR 3.3.4.5	≥ ~~[45.7]~~ %
c. Steam Generator Pressure Difference - High (A > B) or (B > A)	1,2,3	SR 3.3.4.1 SR 3.3.4.2[a][b] SR 3.3.4.4[a][b] SR 3.3.4.5	≤ ~~[48.3]~~ psid
d. Feedwater Header Pressure Difference – High	1, 2, 3	SR 3.3.3.1, SR 3.3.3.2 SR3.3.3.4, SR 3.3.3.5	≥ 56.0 and ≤ 157.5 psid

(a) If the as-found channel setpoint is outside its predefined ~~as-found tolerance~~, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left ~~tolerance~~ around the ~~Limiting~~ Trip Setpoint (~~LTSP~~) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the ~~LTSP~~ are acceptable provided that the as-found and as-left ~~tolerances~~ apply to the actual setpoint implemented in the Surveillance procedures ~~(Nominal Trip Setpoint)~~ to confirm channel performance. The ~~LTSP~~ and the methodologies used to determine the as-found and as-left ~~tolerances~~ are specified in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~.

(e) Steam Generator Pressure - Low may be manually bypassed when steam generator pressure is < ~~[785] psia~~. The bypass shall be automatically removed whenever steam generator pressure is ≥ ~~[785] psia~~.

(f) ~~Only the~~ Main Steam Isolation Signal (MSIS) Function ~~and the~~ Steam Generator Pressure - Low and Containment Pressure - High signals are not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed ~~and [de-activated]~~.

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.3, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The heading for ISTS 3.3.4 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2). Therefore, ISTS 3.3.2 is not included in the PSL ITS. The successive Specifications are renumbered as necessary.

4. PSL does not use the terms “as-found tolerance” and “Limiting Trip Setpoint (LTSP)” in plant specific instrument calculations or the Updated Final Safety Analysis Report. Therefore, the use of these terms in Footnotes (a) and (b) in ISTS Table 3.3.1-1 are revised in the ITS to include the plant specific terminology. To comply with the guidance provided in NRC Regulatory Issue Summary 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36, ‘Technical Specifications,’ Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels,” the PSL setpoint calculations have been structured to include a Nominal Trip Setpoint (NTSP), also known as the field trip setpoint, and determination of an OPERABILITY limit range, referred to in CTS as acceptance criteria band. For PSL, the Nominal Trip Setpoint (NTSP) is synonymous with the Limiting Trip Setpoint (LTSP) and the as-found acceptance criteria band (i.e., OPERABILITY limit range) is synonymous with the as-found tolerance.

5. Condition B is added to ITS 3.3.3 (ISTS 3.3.4) to provide specific actions for one inoperable Containment Sump Recirculation Actuation Signal (RAS) trip unit or associated instrument. In addition, Condition C is also added to provide specific actions for one or more inoperable Auxiliary Feedwater Actuation Signal (AFAS) trip unit or associated instrument. A Note is added to proposed ITS 3.3.3 Condition C (Insert 1 to ISTS 3.3.4) stating that Required Action C.2.2.2 shall be completed when Condition C is entered. This Note constitutes an “unless otherwise stated” exception to LCO 3.0.2 to require the action to be completed within the associated Completion Time even when the LCO is no longer applicable. The added Note is similar to other Notes in the ISTS (e.g., Note to Condition A of ISTS 3.2.3 and the Note to Conditions A and C of ISTS 3.4.3.). These additional ITS ACTIONS are consistent with the PSL current licensing basis and consider the differing actions related to RAS, AFAS, and the remaining ESFAS functions based on whether instruments must be restored in 48 hours, may be optionally restored in accordance with the Risk Informed Completion Time Program, or channel restoration is extended to prior to entering MODE 3 following the next MODE 5 entry. Subsequent Conditions and Required Actions have been renumerated, as applicable, and Condition exceptions revised to reflect the additional ITS ACTIONS.

6. **Unit 2 only:** ISTS 3.3.4, Required Action B.2.2 is deleted and the Completion Time of Required Action B.2.1 (ITS 3.3.3, Required Action D.2) is modified in the Unit 2 ITS consistent with the PSL current licensing basis to allow the channel to be in trip or bypass until the next plant startup following entry into MODE 5 and similar to ISTS 3.3.5 (Digital), Required Action A.2. The Unit 2 ESFAS design includes independence of the four protection channels to ensure no single failure (e.g., circuit fault) can adversely affect more than one channel. As discussed in Section 7.2.4 of NUREG-0843, "Safety Evaluation Report – St. Lucie Unit 2," the NRC concluded that the ESFAS design enhancements were sufficient to ensure the independence of the four protection channels. Thus, the Unit 2 Technical Specifications allow a failed channel to be placed in long term trip or bypass condition provided the channel is restored no later than the next cold shutdown following the channel malfunction. A Note is added to ITS 3.3.3 Condition D (ISTS 3.3.4, Condition B) stating that Required Action D.2 shall be completed when Condition D is entered. This Note constitutes an "unless otherwise stated" exception to LCO 3.0.2 to require the action to be completed within the associated Completion Time even when the LCO is no longer applicable. The added Note is similar to other Notes in the ISTS (e.g., Note to Condition A of ISTS 3.2.3 and the Note to Conditions A and C of ISTS 3.4.3.).

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION ~~(Analog)~~ 3

B 3.3.~~4~~ Engineered Safety Features Actuation System (ESFAS) Instrumentation ~~(Analog)~~ 3

3

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary and to mitigate accidents. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the ESFAS, as well as LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSSs for variables that have significant safety functions. LSSS are defined by the regulation as "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

~~REVIEWER'S NOTE~~

~~The term "Limiting Trip Setpoint" [LTSP] is generic terminology for the calculated trip setting (setpoint) value calculated by means of the plant specific setpoint methodology documented in a document controlled under 10 CFR 50.59. The term [LTSP] indicates that no additional margin has been added between the Analytical Limit and the calculated trip setting.~~ 4

~~"Nominal Trip Setpoint [NTSP]" is the suggested terminology for the actual setpoint implemented in the plant surveillance procedures where margin has been added to the calculated [LTSP]. The as-found and as-left tolerances will apply to the [NTSP] implemented in the Surveillance procedures to confirm channel performance.~~ 4

BASES

BACKGROUND (continued)

~~Licensees are to insert the name of the document(s) controlled under 10 CFR 50.59 that contain the methodology for calculating the as-left and as-found tolerances, in Note b of Table 3.3.4-1 for the phrase "[insert the name of a document controlled under 10 CFR 50.59 such as the Technical Requirements Manual or any document incorporated into the facility FSAR]" throughout these Bases.~~

~~Where the [LTSP] is not included in Table 3.3.4-1, the plant specific location for the [LTSP] or [NTSP] much be cited in Note b of Table 3.3.4-1. The brackets indicate plant specific terms may apply, as required and approved by the NRC.~~

The ~~[Limiting~~ Trip Setpoint (~~LTSP)]~~ specified in ~~Table 3.3.4-1~~ is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the ~~[LTSP]~~ accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the ~~[LTSP]~~ ensures that SLs are not exceeded. As such, the ~~[LTSP]~~ meets the definition of an LSSS (Ref. ~~1~~).

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety function(s)." Relying solely on the ~~[LTSP]~~ to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a Surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the ~~[LTSP]~~ due to some drift of the setting may still be OPERABLE because drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the ~~[LTSP]~~ and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE because it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left ~~tolerance~~ around the ~~[LTSP]~~ to account for further drift during the next surveillance interval. Note that,

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BACKGROUND (continued)

although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left ~~tolerance~~, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

However, there is also some point beyond which the channel may not be able to perform its function due to, for example, greater than expected drift. This value needs to be specified in the Technical Specifications in order to define OPERABILITY of the channels and is designated as the Allowable Value.

If the actual setting (as-found setpoint) of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found ~~tolerance~~ band, the channel is OPERABLE, but degraded. The degraded condition will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the ~~[Normal Trip Setpoint (~~NTSP~~)]~~ (within the allowed ~~tolerance~~), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

During anticipated operational occurrences (AOOs), which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling,
- Fuel centerline melting shall not occur, and
- The Reactor Coolant System (RCS) pressure SL of ~~[~~2750~~]~~ psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. ~~2~~) and 10 CFR ~~100~~ (Ref. ~~3~~) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~ (Ref. ~~3~~) limits. Different accident categories allow

BASES

BACKGROUND (continued)

a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The ESFAS contains devices and circuitry that generate the following signals when the monitored variables reach levels that are indicative of conditions requiring protective action:

1. Safety Injection Actuation Signal (SIAS),
2. Containment Spray Actuation Signal (CSAS),
3. Containment Isolation Signal (CI~~A~~S),
4. Main Steam Isolation Signal (MSIS),
5. Recirculation Actuation Signal (RAS), and
6. Auxiliary Feedwater Actuation Signal (AFAS).

Equipment actuated by each of the above signals is identified in the FSAR (Ref. ~~4~~).

Each of the above ESFAS actuation systems is segmented into four sensor subsystems and two actuation subsystems. Each sensor subsystem includes measurement channels and bistables. ~~The actuation subsystems include two logic subsystems for sequentially loading the diesel generators.~~

Each of the four sensor subsystem channels monitors redundant and independent process measurement channels. Each sensor is monitored by at least one bistable. The bistable associated with each ESFAS Function will trip when the monitored variable exceeds the ~~[LTSP]~~. When tripped, the sensor subsystems provide outputs to the two actuation subsystems.

The two independent actuation subsystems compare the four sensor subsystem outputs. If a trip occurs in the same parameter in two or more sensor subsystem channels, the two-out-of-four logic in each actuation subsystem will initiate one train of ESFAS. Each train can provide protection to the public in the case of a Design Basis Event. Actuation Logic is addressed in LCO 3.3.~~5~~, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual ~~Trip~~."

Each of the four sensor subsystems is mounted in a separate cabinet, excluding the sensors and field wiring.

BASES

BACKGROUND (continued)

The role of the sensor subsystem (measurement channels and bistables) is discussed below; actuation subsystems are discussed in LCO 3.3.~~5~~ 4. 1

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

Four identical measurement channels with electrical and physical separation are provided for each parameter used in the generation of trip signals. These are designated Channels A through D. Measurement channels provide input to ESFAS bistables within the same ESFAS channel. ~~In addition, some measurement channels may also be used as inputs to Reactor Protective System (RPS) bistables, and most provide indication in the control room.~~ Measurement channels used as an input to ~~the RPS or~~ ESFAS are not used for control Functions. 1

When a channel monitoring a parameter indicates an unsafe condition, the bistable monitoring the parameter in that channel will trip. Tripping two or more channels of bistables monitoring the same parameter will de-energize both channels of Actuation Logic of the associated Engineered Safety Features (ESF) equipment.

Three of the four measurement and bistable channels are necessary to meet the redundancy and testability of GDC 21 in Appendix A to 10 CFR 50 (Ref. 2). The fourth channel provides additional flexibility by allowing one channel to be removed from service (trip channel bypass) for maintenance or testing while still maintaining a minimum two-out-of-three logic. The single failure criterion is met during this condition. 1

~~In order to take full advantage of the four channel design, adequate channel to channel independence must be demonstrated, and approved by the NRC staff. Plants not currently licensed to credit four channel independence that may desire this capability must have approval of the NRC staff documented by an NRC Safety Evaluation Report (Ref. 5). Adequate channel to channel independence includes physical and electrical independence of each channel from the others. Furthermore, each channel must be energized from separate inverters and station batteries. Plants not demonstrating four channel independence may operate in a two-out-of three logic configuration for 48 hours.~~ 1

Since no single failure will either cause or prevent a protective system actuation and no protective channel feeds a control channel, this arrangement meets the requirements of IEEE Standard ~~79~~279-1971 (Ref. 6). 1 3

BASES

BACKGROUND (continued)

Bistable Trip Units

Bistable trip units receive an analog input from the measurement channels, compare the analog input to trip setpoints, and provide contact output to the Actuation Logic. They also provide local trip indication and remote annunciation.

There are four channels of bistables, designated A through D, for each ESF Function, one for each measurement channel. ~~In cases where two ESF Functions share the same input and trip setpoint (e.g., containment pressure input to CSAS, CIAS, and SIAS and a Pressurizer Pressure - Low input to the RPS and SIAS), the same bistable may be used to satisfy both Functions.~~

The trip setpoints and Allowable Values used in the bistables are based on the analytical limits stated in Reference 7. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment effects, for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 8), Allowable Values specified in Table 3.3.~~4~~-1, in the accompanying LCO, are conservatively adjusted with respect to the analytical limits. A detailed description of the method used to calculate the trip setpoints, including their explicit uncertainties, is provided in ~~the "Plant Protection System Selection of Trip Setpoint Values"~~ (Ref. 9). The actual trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the interval between surveillances.

The ~~[LTSP]~~ is the value at which the bistable is set and is the expected value to be achieved during calibration. The ~~[LTSP]~~ value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties.

~~[LTSPs]~~, in conjunction with the use of as-found and as-left ~~tolerances~~, consistent with the requirements of the Allowable Value will ensure that Safety Limits of Chapter 2.0, "SAFETY LIMITS (SLs)," are not violated during AOOs and that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

BASES

BACKGROUND (continued)

Note that in the accompanying LCO 3.3.~~4~~, the Allowable Values of the Table 3.3.~~4~~-1 are the least conservative value of the as-found setpoint that a channel can have during a periodic CHANNEL CALIBRATION or CHANNEL FUNCTIONAL TEST.

ESFAS Logic

It is possible to change the two-out-of-four ESFAS logic to a two-out-of-three logic for a given input parameter in one channel at a time by ~~disabling~~ bypassing one channel input to the logic. Thus, the bistables will function normally, producing normal trip indication and annunciation~~, but ESFAS actuation will not occur since the bypassed channel is effectively removed from the coincidence logic. Trip channel bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. At some plants an interlock prevents simultaneous trip channel bypassing of the same parameter in more than one channel.~~ Trip channel bypassing is normally employed during maintenance or testing.

ESFAS Logic is addressed in LCO 3.3.~~5~~.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. Functions such as Manual Initiation, not specifically credited in the accident analysis, serve as backups to Functions and are part of the NRC approved licensing basis for the plant.

~~Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.~~

ESFAS protective Functions are as follows:

APPLICABLE SAFETY ANALYSES (continued)

1. Safety Injection Actuation Signal

The SIAS ensures acceptable consequences during loss of coolant accident (LOCA) events, including steam generator tube rupture, and main steam line breaks (MSLBs) or feedwater line breaks (FWLBs) (inside containment). To provide the required protection, either a high containment pressure or a low pressurizer pressure signal will initiate SIAS. SIAS initiates the Emergency Core Cooling Systems (ECCS), control room isolation, and several other Functions, such as starting the emergency diesel generators.

2. Containment Spray Actuation Signal

The CSAS initiates containment spray, preventing containment overpressurization during a LOCA or MSLB. ~~At some plants,~~ both a high- high containment pressure signal and an SIAS have to actuate to provide the required protection. This configuration reduces the likelihood of inadvertent containment spray. 1

3. Containment Isolation Signal 1

The CIS actuates the Containment Isolation System, ensuring acceptable consequences during LOCAs and MSLBs or FWLBs (inside containment). To provide protection, a high containment pressure signal will initiate CIS at the same setpoint at which an SIAS is generated. A high containment radiation signal will also initiate CIS. 1

4. Main Steam Isolation Signal

The MSIS ensures acceptable consequences during an MSLB or FWLB by isolating both steam generators if either generator indicates a low steam generator pressure. The MSIS, concurrent with or following a reactor trip, minimizes the rate of heat extraction and subsequent cooldown of the RCS during these events. 1

5. Recirculation Actuation Signal

At the end of the injection phase of a LOCA, the refueling water tank (RWT) will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from RWT to the containment sump must occur before the RWT empties to prevent damage to the ECCS pumps and

APPLICABLE SAFETY ANALYSES (continued)

a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support pump suction. Furthermore, early switchover must not occur to ensure sufficient borated water is injected from the RWT to ensure the reactor remains shut down in the recirculation mode. An RWT Level - Low signal initiates the RAS.

6. Auxiliary Feedwater Actuation Signal

An AFAS initiates feedwater flow to ~~both~~ steam generator~~s~~ if a low level is indicated in ~~either~~ steam generator, unless the generator is ~~ruptured~~. 1

the affected

faulted

the affected

The AFAS maintains a steam generator heat sink during the following events:

- MSLB,
- FWLB,
- Inadvertent opening of a steam generator atmospheric dump valve, and
- Loss of feedwater.

A low steam generator water level signal will initiate auxiliary feed to the affected steam generator.

, or feedwater header (FW) differential pressure (FW-A > FW-B) or (FW-B > FW-A), 1

Secondary steam generator (SG) differential pressure (SG-A > SG-B) or (SG-B > SG-A) inhibits auxiliary feed to a generator identified as being ~~ruptured~~. This input to the AFAS logic prevents loss of the intact generator while preventing feeding a ~~ruptured~~ generator during MSLBs and FWLBs. This prevents containment overpressurization during these events. 1

faulted

faulted

The ESFAS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires all channel components necessary to provide an ESFAS actuation to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The specific criteria for determining channel OPERABILITY differ slightly between Functions. These criteria are discussed on a Function by Function basis below.

BASES

LCO (continued)

Actions allow maintenance (trip channel) bypass of individual channels~~, but the bypass activates interlocks that prevent operation with a second channel in the same Function bypassed~~. Plants are restricted to 48 hours in a trip channel bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic). ~~At plants where adequate channel to channel independence has been demonstrated, specific exceptions may be approved by the NRC staff to permit one of the two-out-of-four channels to be bypassed for an extended period of time.~~

Allowable Values for ESFAS Instrumentation (Analog) Functions are specified in Table 3.3.~~4~~-1. ~~[LTSPs]~~ and the methodologies for calculation of the as-left and as-found ~~tolerances~~ are described in ~~[insert the name of a document controlled under 10 CFR 50.59 such as~~ the Technical Requirements Manual~~or any document incorporated into the facility FSAR]~~. The ~~[LTSPs]~~ are selected to ensure that the actual setpoints remain conservative with respect to the as-found ~~tolerance~~ band between successive CHANNEL CALIBRATIONS. After each calibration the trip setpoint shall be left within the as-left band around the ~~[LTSP]~~.

The Bases for the LCO on ESFAS Functions are:

1. Safety Injection Actuation Signal

   a. Containment Pressure - High

      This LCO requires four channels of SIAS Containment Pressure - High to be OPERABLE in MODES 1, 2, and 3.

      The Allowable Value for this trip is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an offnormal condition. The setting is low enough to initiate the ESF Functions when an offnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

   b. Pressurizer Pressure - Low

      This LCO requires four channels of SIAS Pressurizer Pressure - Low to be OPERABLE in MODES 1, 2, and 3.

LCO (continued)

The Allowable Value for this trip is set low enough to prevent actuating the SIAS during normal plant operation and pressurizer pressure transients. The setting is high enough that with a LOCA or MSLB it will actuate to perform as expected, mitigating the consequences of the accidents.

The Pressurizer Pressure - Low trip may be blocked when pressurizer pressure is reduced during controlled plant shutdowns. This block is permitted below ~~1800~~ 1725 psia, and block permissive responses are annunciated in the control room. This allows for a controlled depressurization of the RCS, while maintaining administrative control of ESF protection. From a blocked condition, the block will be automatically removed as pressurizer pressure increases above ~~1800~~ 1725 psia, as sensed by two of the four sensor subsystems~~, in accordance with the bypass philosophy of removing bypasses when the enabling conditions are no longer satisfied~~.

This LCO requires four channels of the bypass permissive removal for SIAS Pressurizer Pressure - Low to be OPERABLE in MODES 1, 2, and 3.

The bypass permissive channels consist of four sensor subsystems and two actuation subsystems. This LCO applies to failures in the four sensor subsystems, including sensors, bistables, and associated equipment. Failures in the actuation subsystems, including the manual bypass key switches, are considered Actuation Logic failures and are addressed in LCO 3.3.~~5~~ 4.

This LCO applies to the bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue.

The block permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow blocking prior to reaching the trip setpoint.

2. Containment Spray Actuation Signal

CSAS is initiated either manually or automatically. ~~At many plants,~~ it is also necessary to have an automatic or manual SIAS for complete actuation. ~~The SIAS opens the containment spray valves, whereas the CSAS actuates other required components.~~ The SIAS requirement should always be satisfied on a legitimate CSAS, since

BASES

LCO (continued)

lower than the Containment Pressure – High-High signal

The parameters used for CSAS (Containment Pressure High-High and SIAS) give direct indication of a LOCA. The time to actuate CSAS depends on the break size. The Containment Pressure High-High set point is the controlling parameter since for all break sizes it takes longer to reach this set point than the SIAS set point.

the Containment Pressure - High signal setpoint used in the SIAS is ~~the same~~ setpoint used in the CSAS. ~~At many plants, the transmitters used to initiate CSAS are independent of those used in the SIAS to prevent inadvertent containment spray due to failures in two sensor channels.~~ 1

a. Containment Pressure - High -High 1

This LCO requires four channels of CSAS Containment Pressure - High to be OPERABLE in MODES 1, 2, and 3. -High 1

The Allowable Value is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an offnormal condition. The setting is low enough to initiate the ESF Functions when an offnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

~~The Containment Pressure - High setpoint is the same in the SIAS (Function 1), CSAS (Function 2), and CIAS (Function 3). However, different sensors and logic are used in each of these Functions.~~ 1

3. Containment Isolation ~~Actuation~~ Signal 1

a. Containment Pressure - High

This LCO requires four channels of CI~~A~~S Containment Pressure - High to be OPERABLE in MODES 1, 2, and 3. 1

The Allowable Value is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an offnormal condition. The setting is low enough to initiate the ESF Functions when an offnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

The Containment Pressure - High setpoint is the same in the SIAS (Function 1)~~, CSAS (Function 2),~~ and CI~~A~~S (Function 3). However, different ~~sensors and~~ logic ~~are~~ is used in each of these Functions. 1

LCO (continued)

b. Containment Radiation - High

This LCO requires four channels of CI~~A~~S Containment Radiation - High to be OPERABLE in MODES 1, 2, and 3. 1

The Allowable Value is high enough to avoid unnecessary actuation, but adequate to provide diverse actuation of the CI~~A~~S in the event of a LOCA. 1

4. Main Steam Isolation Signal

The MSIS is required to be OPERABLE in MODES 1, 2, and 3 except when all associated valves are closed and de-activated.

a. Steam Generator Pressure - Low

This LCO requires four channels of MSIS Steam Generator Pressure - Low for each steam generator to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value is set below the full load operating value for steam pressure so as not to interfere with normal plant operation. However, the setting is high enough to provide the required protection for excessive steam demand. An excessive steam demand causes the RCS to cool down, resulting in a positive reactivity addition to the core. An MSIS is required to prevent the excessive cooldown.

This Function may be manually blocked when steam generator pressure is reduced during controlled plant cooldowns. The block is permitted below ~~785~~ psia, and block permissive responses are annunciated in the control room. This allows a controlled depressurization of the secondary system, while maintaining administrative control of ESF protection. From a blocked condition, the block will be removed automatically as steam generator pressure increases above ~~785~~ psia, as sensed by two of the four sensor subsystems~~, in accordance with the bypass philosophy of removing bypasses when the enabling conditions are no longer satisfied~~.

685 685 1 1 1

This LCO requires four channels per steam generator of the bypass removal for MSIS Steam Generator Pressure - Low to be OPERABLE in MODES 1, 2, and 3.

1 3 1

LCO (continued)

The bypass removal channels consist of four sensor subsystems and two actuation subsystems. This LCO applies to failures in the four sensor subsystems, including sensors, bistables, and associated equipment. Failures in the actuation subsystems, including the manual bypass key switches, are considered Actuation Logic failures and are addressed in LCO 3.3.~~5~~.4 3

This LCO applies to the bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue.

The block permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow blocking prior to reaching the trip setpoint.

5. Recirculation Actuation Signal

a. Refueling Water Tank Level - Low

This LCO requires four channels of RWT Level - Low to be OPERABLE in MODES 1, 2, and 3.

The upper limit on the Allowable Value for this trip is set low enough to ensure RAS does not initiate before sufficient water is transferred to the containment sump. Premature recirculation could impair the reactivity control Function of safety injection by limiting the amount of boron injection. Premature recirculation could also damage or disable the recirculation system if recirculation begins before the sump has enough water to prevent air ~~containment~~ entrainment in the suction. The lower limit on the RWT Level - Low trip Allowable Value is high enough to transfer suction to the containment sump prior to emptying the RWT. 1

6. Auxiliary Feedwater Actuation Signal

The AFAS logic actuates auxiliary feedwater (AFW) to a steam generator on low level in that generator unless it has been identified as being ~~ruptured~~ faulted.

A low level in ~~either~~ the faulted steam generator, as sensed by each steam a two-out-of-four coincidence of four wide range sensors for ~~any~~ generator, will generate an AFAS start signal, which starts ~~both~~ the affected trains of AFW 1 1

BASES

LCO (continued)

pumps and feeds ~~both~~ steam generator~~s~~. The AFAS also monitors the secondary differential pressure in both steam generators and initiates an AFAS block signal to a ~~ruptured~~ generator, if the pressure in ~~that generator is lower than that in the other generator~~ by the differential pressure setpoint.

a. Steam Generator A/B Level - Low

This LCO requires four channels for each steam generator of Steam Generator Level - Low to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value ensures adequate time exists to initiate AFW while the steam generators can function as a heat sink.

b. Steam Generator Pressure Difference - High (SG-A > SG-B) or (SG-B > SG-A)

This LCO requires four channels per steam generator of Steam Generator Pressure Difference - High to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value for this trip is high enough to allow for small pressure differences and normal instrumentation errors between the steam generator channels during normal operation without an actuation. The setting is low enough to detect and inhibit feeding of a ~~ruptured~~ steam generator in the event of an MSLB or FWLB, while permitting the feeding of the intact steam generator.

INSERT 1

The ESFAS channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

APPLICABILITY All ESFAS Functions are required to be OPERABLE in MODES 1, 2, and 3. In MODES 1, 2, and 3 there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the main steam isolation valves to preclude a positive reactivity addition,
- Actuate AFW to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available),
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or MSLB, and

1

INSERT 1

b. Feedwater Header Pressure Difference - High
(FW-A > FW-B) or (FW-B > FW-A)

This LCO requires four channels per feedwater subsystem of Feedwater Header Pressure Difference - High to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value for this trip is high enough to allow for small pressure differences and normal instrumentation errors between the feedwater header channels during normal operation without an actuation. The setting is low enough to detect and inhibit feeding of a faulted steam generator in the event of an MSLB or FWLB, while permitting the feeding of the intact steam generator.

BASES

APPLICABILITY (continued)

- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

In MODES 4, 5, and 6, automatic actuation of ESFAS Functions is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating the ESF components, if required, as addressed by LCO 3.3.~~5~~. In LCO 3.3.~~5~~, manual capability is required for Functions other than AFAS in MODE 4, even though automatic actuation is not required. Because of the large number of components actuated on each ESFAS, actuation is simplified by the use of the Manual ~~Trip~~ push buttons. Manual ~~Trip~~ of AFAS is not required in MODE 4 because AFW or shutdown cooling will already be in operation in this MODE.

Actuation

Actuation

The ESFAS Actuation Logic must be OPERABLE in the same MODES as the automatic and Manual ~~Trip~~. In MODE 4, only the portion of the ESFAS logic responsible for the required Manual ~~Trip~~ must be OPERABLE.

Actuation

Actuation

In MODES 5 and 6, ESFAS initiated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis.

Typically, the drift is small and results in a delay of actuation rather than a total loss of function. Determination of setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the actual trip setpoint is nonconservative with respect to the Allowable Value in Table 3.3.~~4~~-1, the channel is inoperable and the appropriate Condition(s) are entered.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value in Table 3.3.~~4~~-1, or the channel is not functioning as required, or the sensor, instrument loop, signal processing electronics, or ESFAS bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the plant must enter the Condition statement for the particular protection Function affected.

BASES

ACTIONS (continued)

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A Note has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.~~4~~-1. Completion Times for the inoperable channel of a Function will be tracked separately.

INSERT A

~~[A.1~~

INSERT B

INSERT C

~~Condition A applies to one CSAS Containment Pressure - High channel inoperable. CSAS logic is identical to that of the other ESFAS Functions; however, the inadvertent actuation of a CSAS is undesirable, since it may damage equipment inside containment. For this reason, placing the inoperable channel in trip is not an option as it is in Conditions B and C.]~~

~~[For those plants in which the SIAS is required for a complete CSAS actuation, Condition B for one ESFAS channel inoperable and Condition C for two ESFAS channels inoperable may be preferable to Condition A.~~

~~If one CSAS channel is inoperable, operation is allowed to continue, providing the inoperable channel is placed in bypass within 1 hour. The Completion Time of 1 hour allotted to bypass the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.]~~

~~B~~.1, ~~B~~.2.1, and ~~B~~.2.2

Condition ~~B~~ applies to the failure of a single channel of one or more input parameters in the following ESFAS Functions:

1. Safety Injection Actuation Signal
   Containment Pressure - High
   Pressurizer Pressure – Low

2. Containment Spray Actuation Signal
   Containment Pressure - High-High

3. Containment Isolation Actuation Signal
   Containment Pressure - High
   Containment Radiation - High

1

INSERT A

A.1, A.2.1, and A.2.2

Condition A applies to the failure of a single channel of one or more input parameters in the Containment Spray Actuation Signal (CSAS) ESFAS Function.

CSAS coincidence logic is normally two-out-of-four. If one CSAS channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one CSAS channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action A.1). With one channel in bypass, no additional random failure of a single channel could spuriously cause an actuation and a valid trip signal can still cause an actuation. With one channel in trip, an additional random failure of a single channel could spuriously cause an actuation. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to bypass or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

One failed channel is restored to OPERABLE status or is placed in trip within 48 hours (Required Action A.2.1 or A.2.2). Required Action A.2.1 restores the full capability of the function. Required Action A.2.2 places the function in a one-out-of-three configuration. In this configuration, common cause failure of the dependent channel cannot prevent ESFAS actuation. The 48 hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 48 hour period is a low probability event.

1

INSERT B

B.1 and B.2

Condition B applies to the failure of a single channel of one or more input parameters in the Containment Sump Recirculation Actuation Signal (RAS) ESFAS Function.

ESFAS coincidence logic is normally two-out-of-four. If one ESFAS channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one ESFAS channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action B.1). With one channel in bypass, no additional random failure of a single channel could spuriously cause an actuation and a valid trip signal can still cause an actuation. With one channel in trip, an additional random failure of a single channel could spuriously cause an actuation. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to bypass or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

The failed channel is restored to OPERABLE status within 48 hours (Required Action B.2). Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program (Required Action B.2). The 48 hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 48 hour period is a low probability event.

1

INSERT C

C.1, C.2.1, C.2.2.1, C.2.2.2

Condition C applies to the failure of a single channel of one or more input parameters in the Auxiliary Feedwater Actuation Signal (AFAS) ESFAS Function.

AFAS coincidence logic is normally two-out-of-four. If one AFAS channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one AFAS channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action C.1). With one channel in bypass, no additional random failure of a single channel could spuriously cause an actuation and a valid trip signal can still cause an actuation. With one channel in trip, an additional random failure of a single channel could spuriously cause an actuation. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to bypass or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

One failed channel is restored to OPERABLE status or is placed in bypass within 48 hours (Required Action C.2.1 or C.2.2.1). Required Action C.2.1 restores the full capability of the function. Required Action C.2.2.1 places the function in a two-out-of-three configuration. In this configuration, common cause failure of the dependent channel cannot prevent AFAS actuation. The 48 hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 48 hour period is a low probability event. R1

Additionally, the failed channel must be restored to OPERABLE status prior to entering MODE 3 following the next MODE 5 entry. The Completion Time of Required Action C.2.2.2 is based on adequate channel to channel independence, which allows a two-out-of-three channel operation since no single failure will cause or prevent a reactor trip. However, it is expected that the inoperable channel will be restored to OPERABLE status at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying restoration of the channel as well as any plant configuration changes required or shutting the plant down to repair the channel) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to repair the channel. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and NRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

Condition C is modified by a Note requiring Required Action C.2.2.2 to be completed whenever the Condition is entered even when the LCO is no longer applicable to ensure the channel is restored to OPERABLE prior to the next reactor startup from MODE 5 conditions.

BASES

ACTIONS (continued)

4. Main Steam Isolation Signal
   Steam Generator Pressure - Low

5. ~~Recirculation Actuation Signal~~
   ~~Refueling Water Tank Level - Low~~

6. ~~Auxiliary Feedwater Actuation Signal~~
   ~~Steam Generator Level - Low~~
   ~~Steam Generator Pressure Difference - High~~

ESFAS coincidence logic is normally two-out-of-four. If one ESFAS channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one ESFAS channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action ~~B~~.1). With one channel in bypass, no additional random failure of a single channel could spuriously ~~trip the reactor~~ and a valid trip signal can still ~~trip the reactor~~. With one channel in trip, an additional random failure of a single channel could spuriously ~~trip the reactor~~. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to bypass or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

One failed channel is restored to OPERABLE status or is placed in trip within ~~[~~48~~]~~ hours ~~[or in accordance with the Risk Informed Completion Time Program]~~ (Required Action ~~B~~.2.1 or ~~B~~.2.2). Required Action ~~B~~.2.1 restores the full capability of the function. Required Action ~~B~~.2.2 places the function in a one-out-of-three configuration. In this configuration, common cause failure of the dependent channel cannot prevent ESFAS actuation. The ~~[~~48~~]~~ hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the ~~[~~48~~]~~ hour period is a low probability event. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~

INSERT E

~~C~~.1 and ~~C~~.2

Condition ~~C~~ applies to the failure of two channels in any of the following ESFAS functions:

1

INSERT E

E.1 and E.2

Condition E applies to failure of two or more channels in the Auxiliary Feedwater Actuation Signal (AFAS) ESFAS Function or two channel in the Containment Spray Actuation System (CSAS) ESAFS Function.

With two inoperable channels, one channel should be placed in bypass, and the other channel should be placed in trip within the 1 hour Completion Time. With one channel of protective instrumentation bypassed, the ESFAS Function is in two-out-of-three logic, but with another channel failed the ESFAS may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESFAS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, ESFAS actuation will occur.

One of the failed channels should be restored to OPERABLE status within 48 hours. The 48 hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 48 hour period is a low probability event. After one channel is restored to OPERABLE status, the provisions of Condition C still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action E.2 must be placed in bypass if more than 48 hours has elapsed since the initial channel failure. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

BASES

ACTIONS (continued)

1. Safety Injection Actuation Signal
   Containment Pressure - High
   Pressurizer Pressure - Low

3. Containment Isolation ~~Actuation~~ Signal
   Containment Pressure - High
   Containment Radiation - High

4. Main Steam Isolation Signal
   Steam Generator Pressure - Low

5. Recirculation Actuation Signal
   Refueling Water Tank Level - Low

~~6. Auxiliary Feedwater Actuation Signal~~
~~Steam Generator Level - Low~~
~~Steam Generator Pressure Difference - High~~

With two inoperable channels, one channel should be placed in bypass, and the other channel should be placed in trip within the 1 hour Completion Time. With one channel of protective instrumentation bypassed, the ESFAS Function is in two-out-of-three logic, but with another channel failed the ESFAS may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESFAS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, ESFAS actuation will occur.

One of the failed channels should be restored to OPERABLE status within ~~[~~48~~]~~ hours, for reasons similar to those stated under Condition ~~B~~. After one channel is restored to OPERABLE status, the provisions of Condition ~~B~~ still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action ~~C~~.2 must be placed in trip if more than ~~[~~48~~]~~ hours has elapsed since the initial channel failure.

except for Function 5 - RAS

Function 1 - SIAS, Function 3 – CIS, Function 4 – MSIS

After one Function 5 – RAS channel is restored to OPERABLE status, the provisions of Condition B still apply to the remaining inoperable channel. The RAS channel that is still inoperable after completion of Required Action F.2 must be restored to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time since the initial channel failure.

~~D~~.1, ~~D~~.2.1, ~~D~~.2.2.1, and ~~D~~.2.2.2

Condition ~~D~~ applies to the failure of one bypass removal channel.

The bypass removal channels consist of four sensor subsystems and two actuation subsystems. Condition ~~D~~ applies to failures in one of the four sensor subsystems, including sensors, bistables, and associated

BASES

ACTIONS (continued)

equipment. Failures in the actuation subsystems, including the manual bypass key switches, are considered Actuation Logic failures and are addressed in LCO 3.3.~~5~~.

In Condition ~~D~~, it is permissible to continue operation with one bypass permissive removal channel failed, providing the bypass is disabled (Required Action ~~D~~.1). This can be accomplished by removing the bypass with the manual bypass key switch, which disables the bypass in its respective ~~both~~ train~~s~~. Since the bypass Function must be manually enabled, the bypass permissive Function will not by itself cause an undesired bypass insertion.

Alternatively, the bypass may be disabled by defeating the bypass permissive input in one of the four channels to the two-out-of-four bypass removal logic, placing the bypass removal feature in one-out-of-three logic. Thus, any of the remaining three channels is capable of removing the bypass feature when the bypass enable conditions are no longer valid.

If the bypass removal feature in the inoperable channel cannot be defeated, actions to address the inoperability of the affected automatic trip channel must be taken. Required Action ~~D~~.2.1, Required Action ~~D~~.2.2.1, and Required Action ~~D~~.2.2.2 are equivalent to the Required Actions for a single automatic trip channel failure (Condition ~~B~~). The Completion Times have the same bases as discussed for Condition ~~B~~. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~

~~E~~.1, ~~E~~.2.1, and ~~E~~.2.2

Condition ~~E~~ applies to two inoperable bypass removal channels. The bypass removal channels consist of four sensor subsystems and two actuation subsystems. This Condition applies to failures in two of the four sensor subsystems. With two of the four sensor subsystems failed in a nonconservative direction (enabling the bypass Function), the bypass removal feature is in two-out-of-two logic. Failures in the actuation subsystems, including the manual bypass key switches, are considered Actuation Logic failures and are addressed in LCO 3.3.~~5~~.

In Condition ~~E~~, it is permissible to continue operation with two bypass permissive channels failed, providing the bypasses are disabled in a similar manner as discussed for Condition ~~D~~.

BASES

ACTIONS (continued)

If the failed bypasses cannot be disabled, actions to address the inoperability of the affected automatic trip channels must be taken. Required Action ~~E~~.2.1 and Required Action ~~E~~.2.2 are equivalent to the Required Actions for a two automatic trip channel failure (Condition ~~C~~). Also similar to Condition ~~C~~, after one set of inoperable channels is restored, the provisions of Condition ~~D~~ still apply to the remaining inoperable channel, with the Completion Time measured from the point of the initial bypass channel failure. The Completion Times have the same bases as discussed for Condition ~~C~~. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~

~~F~~.1 and ~~F~~.2

If the Required Actions and associated Completion Times of Condition A, B, C, D, ~~or~~ E are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within ~~[~~12~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for any particular ESFAS Function are found in the SRs column of Table 3.3.~~4~~-1 for that Function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and response time testing.

~~REVIEWER'S NOTE~~

~~In order for a unit to take credit for topical reports as the basis for justifying Frequencies, topical reports should be supported by an NRC staff Safety Evaluation Report that establishes the acceptability of each topical report for that unit.~~

~~REVIEWER'S NOTE~~

~~Notes a and b are applied to the setpoint verification Surveillances for each ESFAS Instrumentation (Analog) Function in Table 3.3.4-1 unless one or more of the following exclusions apply:~~

~~1. Manual actuation circuits, automatic actuation logic circuits or instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.~~

~~2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.~~ 4

~~3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.~~

3

SR 3.3.~~4~~.1 3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

~~[The Frequency of about once every shift is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~CHANNEL CHECK supplements less formal, but more frequent, checks of CHANNEL OPERABILITY during normal operational use of displays associated with the LCO required channels.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

3

SR 3.3.~~4~~.2 3

~~[~~ ~~A CHANNEL FUNCTIONAL TEST is performed every [92] days to ensure the entire channel will perform its intended function when needed.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Move to end of this SR 3.3.3.2 description

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

A CHANNEL FUNCTIONAL TEST on each ESFAS instrument channel is performed to ensure the entire channel will perform its intended function when needed.

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. 1

The CHANNEL FUNCTIONAL TEST tests the individual sensor subsystems using an analog test input to each bistable.

BASES

SURVEILLANCE REQUIREMENTS (continued)

A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified ~~tolerance~~ around the setpoint. Any setpoint adjustment shall be consistent with the assumptions of the plant specific setpoint analysis.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. ~~The requirements for this review are outlined in Reference [10].~~

SR 3.3.~~4~~.2 is modified by two Notes as identified in Table 3.3.~~4~~-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[~~LTSP~~]~~. Where a setpoint more conservative than the ~~[~~LTSP~~]~~ is used in the plant surveillance procedures ~~[~~NTSP~~]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[~~LTSP~~]~~, then the channel shall be declared inoperable.

The second Note also requires that ~~[~~LTSP~~]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~.

SR 3.3.~~4~~.3

SR 3.3.~~4~~.3 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.~~4~~.2, except 3.3.~~4~~.3 is ~~performed within 92 days prior to startup and is~~ only applicable to bypass Functions. These include the Pressurizer Pressure - Low bypass and the MSIS Steam Generator Pressure - Low bypass. A successful test of the required contact(s) of a channel relay may be

BASES

SURVEILLANCE REQUIREMENTS (continued)

performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

~~The CHANNEL FUNCTIONAL TEST for proper operation of the bypass removal Functions is critical during plant heatups because the bypasses may be in place prior to entering MODE 3 but must be removed at the appropriate points during plant startup to enable the ESFAS Function. Consequently, just prior to startup is the appropriate time to verify bypass removal Function OPERABILITY. Once the bypasses are removed, the bypasses must not fail in such a way that the associated ESFAS Function is inappropriately bypassed. This feature is verified by the appropriate ESFAS Function CHANNEL FUNCTIONAL TEST.~~ 1

~~The allowance to conduct this Surveillance within 92 days of startup is based upon the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 11).~~ 1

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 1

3

SR 3.3.~~4~~.4 3

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the extension analysis. ~~The requirements for this review are outlined in Reference [10].~~ 1

~~[The Frequency is based upon the assumption of an [18] month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Move to end of this SR 3.3.3.4 description

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.3.4.4 is modified by two Notes as identified in Table 3.3.4-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[~~LTSP~~]~~. Where a setpoint more conservative than the ~~[~~LTSP~~]~~ is used in the plant surveillance procedures ~~[~~NTSP~~]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and~~/or~~ Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[~~LTSP~~]~~, then the channel shall be declared inoperable.

The second Note also requires that ~~[~~LTSP~~]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~.

SR 3.3.4.5

This Surveillance ensures that the train actuation response times are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. The analysis models the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment in both trains reaches the required functional state (e.g.,

SURVEILLANCE REQUIREMENTS (continued)

pumps at rated discharge pressure, valves in full open or closed position). Response time testing acceptance criteria are included in Reference 5. The test may be performed in one measurement or in overlapping segments, with verification that all components are measured.

~~---------- REVIEWER'S NOTE ----------~~
~~Applicable portions of the following TS Bases are applicable to plants adopting CEOG Topical Report CE NPSD-1167-1, "Elimination of Pressure Sensor Response Time Testing Requirements," and the methodology contained in Attachment 1 to TSTF-569.~~
~~----------~~ 2

Response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. ~~Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements," (Ref. 12) provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the Topical Report. The response time may be verified for components that replace the components that were previously evaluated in Ref. 12 provided that the components have been evaluated in accordance with the NRC approved methodology as discussed in Attachment 1 to TSTF-569, "Methodology to Eliminate Pressure Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing," (Ref. 13). Response time verification for other sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and reverified after maintenance that may adversely affect the sensor response time.~~ 1

~~[ESF RESPONSE TIME tests are conducted on a STAGGERED TEST BASIS of once every [18] months. This results in the interval between successive tests of a given channel of n x 18 months, where n is the number of channels in the Function. Surveillance of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The [18] month STAGGERED TEST BASIS Frequency is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~ 2

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

1. Technical Requirements Manual

REFERENCES 2

~~1~~. Regulatory Guide 1,105, "Setpoints for Safety-Related Instrumentation," Revision 3. 1

3 ~~2~~. 10 CFR 50, Appendix A.

4 ~~3~~. 10 CFR 100. 50.67

5 ~~4~~ U FSAR, Section ~~[~~7.3~~]~~. 2

~~5. NRC Safety Evaluation Report, [Date].~~ 2

6. IEEE Standard 279-1971.

7 U FSAR, Chapter ~~[14]~~. 15 2

8. 10 CFR 50.49.

9. IC-3.17, FPL Setpoint Standard. ~~"Plant Protection System Selection of Trip Setpoint Values."~~ 1

~~10. FSAR, Section [7.2].~~ 2

~~11. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989.~~

~~12. CEOG Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements."~~

~~13.~~ Attachment 1 to TSTF-569, "Methodology to Eliminate Pressure ~~Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing."~~

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~4~~ Engineered Safety Features Actuation System (ESFAS) Instrumentation ~~(Analog)~~

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary and to mitigate accidents. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the ESFAS, as well as LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSSs for variables that have significant safety functions. LSSS are defined by the regulation as "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

~~REVIEWER'S NOTE~~

~~The term "Limiting Trip Setpoint" [LTSP] is generic terminology for the calculated trip setting (setpoint) value calculated by means of the plant specific setpoint methodology documented in a document controlled under 10 CFR 50.59. The term [LTSP] indicates that no additional margin has been added between the Analytical Limit and the calculated trip setting.~~

~~"Nominal Trip Setpoint [NTSP]" is the suggested terminology for the actual setpoint implemented in the plant surveillance procedures where margin has been added to the calculated [LTSP]. The as-found and as-left tolerances will apply to the [NTSP] implemented in the Surveillance procedures to confirm channel performance.~~

BASES

BACKGROUND (continued)

~~Licensees are to insert the name of the document(s) controlled under 10 CFR 50.59 that contain the methodology for calculating the as-left and as-found tolerances, in Note b of Table 3.3.4-1 for the phrase "[insert the name of a document controlled under 10 CFR 50.59 such as the Technical Requirements Manual or any document incorporated into the facility FSAR]" throughout these Bases.~~

~~Where the [LTSP] is not included in Table 3.3.4-1, the plant specific location for the [LTSP] or [NTSP] much be cited in Note b of Table 3.3.4-1. The brackets indicate plant specific terms may apply, as required and approved by the NRC.~~

The ~~[Limiting~~ Trip Setpoint (~~LTSP)]~~ specified in ~~Table 3.3.4-1~~ is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the ~~[LTSP]~~ accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the ~~[LTSP]~~ ensures that SLs are not exceeded. As such, the ~~[LTSP]~~ meets the definition of an LSSS (Ref. ~~1~~).

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety function(s)." Relying solely on the ~~[LTSP]~~ to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protection channel setting during a Surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the ~~[LTSP]~~ due to some drift of the setting may still be OPERABLE because drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the ~~[LTSP]~~ and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE because it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left ~~tolerance~~ around the ~~[LTSP]~~ to account for further drift during the next surveillance interval. Note that,

BASES

BACKGROUND (continued)

although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left ~~tolerance~~, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

However, there is also some point beyond which the channel may not be able to perform its function due to, for example, greater than expected drift. This value needs to be specified in the Technical Specifications in order to define OPERABILITY of the channels and is designated as the Allowable Value.

If the actual setting (as-found setpoint) of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found ~~tolerance~~ band, the channel is OPERABLE, but degraded. The degraded condition will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the ~~[Normal Trip Setpoint (~~NTSP~~)]~~ (within the allowed ~~tolerance~~), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

During anticipated operational occurrences (AOOs), which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling,
- Fuel centerline melting shall not occur, and
- The Reactor Coolant System (RCS) pressure SL of ~~[~~2750~~]~~ psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. ~~2~~) and 10 CFR ~~100~~ (Ref. ~~3~~) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~ (Ref. ~~3~~) limits. Different accident categories allow

BASES

BACKGROUND (continued)

a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The ESFAS contains devices and circuitry that generate the following signals when the monitored variables reach levels that are indicative of conditions requiring protective action:

1. Safety Injection Actuation Signal (SIAS),
2. Containment Spray Actuation Signal (CSAS),
3. Containment Isolation Actuation Signal (CIAS),
4. Main Steam Isolation Signal (MSIS),
5. Recirculation Actuation Signal (RAS), and
6. Auxiliary Feedwater Actuation Signal (AFAS).

Equipment actuated by each of the above signals is identified in the FSAR (Ref. ~~4~~ 5). U 1

Each of the above ESFAS actuation systems is segmented into four sensor subsystems and two actuation subsystems. Each sensor subsystem includes measurement channels and bistables. ~~The actuation subsystems include two logic subsystems for sequentially loading the diesel generators.~~ 1

Each of the four sensor subsystem channels monitors redundant and independent process measurement channels. Each sensor is monitored by at least one bistable. The bistable associated with each ESFAS Function will trip when the monitored variable exceeds the ~~[LTSP]~~. When tripped, the sensor subsystems provide outputs to the two actuation subsystems. 2 NTSP

The two independent actuation subsystems compare the four sensor subsystem outputs. If a trip occurs in the same parameter in two or more sensor subsystem channels, the two-out-of-four logic in each actuation subsystem will initiate one train of ESFAS. Each train can provide protection to the public in the case of a Design Basis Event. Actuation Logic is addressed in LCO 3.3.~~5~~, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual ~~Trip~~." 3 4 Actuation

Each of the four sensor subsystems is mounted in a separate cabinet, excluding the sensors and field wiring.

BASES

BACKGROUND (continued)

The role of the sensor subsystem (measurement channels and bistables) is discussed below; actuation subsystems are discussed in LCO 3.3.~~5~~ 4. 1

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

Four identical measurement channels with electrical and physical separation are provided for each parameter used in the generation of trip signals. These are designated Channels A through D. Measurement channels provide input to ESFAS bistables within the same ESFAS channel. ~~In addition, some measurement channels may also be used as inputs to Reactor Protective System (RPS) bistables, and most provide indication in the control room.~~ Measurement channels used as an input to ~~the RPS or~~ ESFAS are not used for control Functions. 1

When a channel monitoring a parameter indicates an unsafe condition, the bistable monitoring the parameter in that channel will trip. Tripping two or more channels of bistables monitoring the same parameter will de-energize both channels of Actuation Logic of the associated Engineered Safety Features (ESF) equipment.

Three of the four measurement and bistable channels are necessary to meet the redundancy and testability of GDC 21 in Appendix A to 10 CFR 50 (Ref. 2). The fourth channel provides additional flexibility by allowing one channel to be removed from service (trip channel bypass) for maintenance or testing while still maintaining a minimum two-out-of-three logic. The single failure criterion is met during this condition. 1

~~In order to take full advantage of the four channel design, adequate channel to channel independence must be demonstrated, and approved by the NRC staff. Plants not currently licensed to credit four channel independence that may desire this capability must have approval of the NRC staff documented by an NRC Safety Evaluation Report (Ref. 5). Adequate channel to channel independence includes physical and electrical independence of each channel from the others. Furthermore, each channel must be energized from separate inverters and station batteries. Plants not demonstrating four channel independence may operate in a two-out-of three logic configuration for 48 hours.~~ 1

Since no single failure will either cause or prevent a protective system actuation and no protective channel feeds a control channel, this arrangement meets the requirements of IEEE Standard ~~79~~279-1971 (Ref. 6). 1 3 1

BASES

BACKGROUND (continued)

Bistable Trip Units

Bistable trip units receive an analog input from the measurement channels, compare the analog input to trip setpoints, and provide contact output to the Actuation Logic. They also provide local trip indication and remote annunciation.

There are four channels of bistables, designated A through D, for each ESF Function, one for each measurement channel. ~~In cases where two ESF Functions share the same input and trip setpoint (e.g., containment pressure input to CSAS, CIAS, and SIAS and a Pressurizer Pressure - Low input to the RPS and SIAS), the same bistable may be used to satisfy both Functions.~~ 1 R1

SIAS and MSIS Functions share the containment pressure measurement channel inputs and trip setpoints. The bistables are shared to satisfy both Functions.

The trip setpoints and Allowable Values used in the bistables are based on the analytical limits stated in Reference 7. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment effects, for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 8), Allowable Values specified in Table 3.3.~~4~~-1, in the accompanying LCO, are conservatively adjusted with respect to the analytical limits. A detailed description of the method used to calculate the trip setpoints, including their explicit uncertainties, is provided in ~~the "Plant Protection System Selection of Trip Setpoint Values"~~ (Ref. 9). The actual trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the interval between surveillances. 3 3 1

IC-3.17, “FPL Setpoint Standard”

NTSP NTSP

The ~~[~~LTSP~~]~~ is the value at which the bistable is set and is the expected value to be achieved during calibration. The ~~[~~LTSP~~]~~ value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. 2

acceptance criteria bands

NTSP ~~[~~LTSPs~~]~~, in conjunction with the use of as-found and as-left ~~tolerances~~, consistent with the requirements of the Allowable Value will ensure that Safety Limits of Chapter 2.0, "SAFETY LIMITS (SLs)," are not violated during AOOs and that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed. 2 1

BASES

BACKGROUND (continued)

Note that in the accompanying LCO 3.3.~~4~~, the Allowable Values of the Table 3.3.~~4~~-1 are the least conservative value of the as-found setpoint that a channel can have during a periodic CHANNEL CALIBRATION or CHANNEL FUNCTIONAL TEST.

ESFAS Logic

It is possible to change the two-out-of-four ESFAS logic to a two-out-of-three logic for a given input parameter in one channel at a time by ~~disabling~~ bypassing one channel input to the logic. Thus, the bistables will function normally, producing normal trip indication and annunciation~~, but ESFAS actuation will not occur since the bypassed channel is effectively removed from the coincidence logic. Trip channel bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. At some plants an interlock prevents simultaneous trip channel bypassing of the same parameter in more than one channel.~~ Trip channel bypassing is normally employed during maintenance or testing.

ESFAS Logic is addressed in LCO 3.3.~~5~~.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. Functions such as Manual Initiation, not specifically credited in the accident analysis, serve as backups to Functions and are part of the NRC approved licensing basis for the plant.

~~Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.~~

ESFAS protective Functions are as follows:

BASES

APPLICABLE SAFETY ANALYSES (continued)

1. Safety Injection Actuation Signal

The SIAS ensures acceptable consequences during loss of coolant accident (LOCA) events, including steam generator tube rupture, and main steam line breaks (MSLBs) or feedwater line breaks (FWLBs) (inside containment). To provide the required protection, either a high containment pressure or a low pressurizer pressure signal will initiate SIAS. SIAS initiates the Emergency Core Cooling Systems (ECCS), control room isolation, and several other Functions, such as starting the emergency diesel generators.

2. Containment Spray Actuation Signal

The CSAS initiates containment spray, preventing containment overpressurization during a LOCA or MSLB. ~~At some plants,~~ both a high- high containment pressure signal and an SIAS have to actuate to provide the required protection. This configuration reduces the likelihood of inadvertent containment spray. 1

3. Containment Isolation Actuation Signal 1

The CI~~A~~S actuates the Containment Isolation System, ensuring acceptable consequences during LOCAs and MSLBs or FWLBs (inside containment). To provide protection, a high containment pressure signal will initiate CIAS at the same setpoint at which an SIAS is generated. A high containment radiation signal will also initiate CIAS. 1

4. Main Steam Isolation Signal

The MSIS ensures acceptable consequences during an MSLB or FWLB by isolating both steam generators if either generator indicates a low steam generator pressure. The MSIS, concurrent with or following a reactor trip, minimizes the rate of heat extraction and subsequent cooldown of the RCS during these events. A high containment pressure signal will also initiate MSIS. 1

5. Recirculation Actuation Signal

At the end of the injection phase of a LOCA, the refueling water tank (RWT) will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from RWT to the containment sump must occur before the RWT empties to prevent damage to the ECCS pumps and

APPLICABLE SAFETY ANALYSES (continued)

a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support pump suction. Furthermore, early switchover must not occur to ensure sufficient borated water is injected from the RWT to ensure the reactor remains shut down in the recirculation mode. An RWT Level - Low signal initiates the RAS.

6. Auxiliary Feedwater Actuation Signal

the affected

An AFAS initiates feedwater flow to ~~both~~ steam generator~~s~~ if a low level is indicated in ~~either~~ steam generator, unless the generator is ~~ruptured~~. 1

faulted

the affected

The AFAS maintains a steam generator heat sink during the following events:

- MSLB,
- FWLB,
- Inadvertent opening of a steam generator atmospheric dump valve, and
- Loss of feedwater.

A low steam generator water level signal will initiate auxiliary feed to the affected steam generator.

, or feedwater header (FW) differential pressure (FW-A > FW-B) or (FW-B > FW-A), 1

Secondary steam generator (SG) differential pressure (SG-A > SG-B) or (SG-B > SG-A) inhibits auxiliary feed to a generator identified as being ~~ruptured~~. This input to the AFAS logic prevents loss of the intact generator while preventing feeding a ~~ruptured~~ generator during MSLBs and FWLBs. This prevents containment overpressurization during these events. 1

faulted

faulted

The ESFAS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires all channel components necessary to provide an ESFAS actuation to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The specific criteria for determining channel OPERABILITY differ slightly between Functions. These criteria are discussed on a Function by Function basis below.

BASES

LCO (continued)

Actions allow maintenance (trip channel) bypass of individual channels~~, but the bypass activates interlocks that prevent operation with a second channel in the same Function bypassed~~. Plants are restricted to 48 hours in a trip channel bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic). ~~At plants where adequate channel to channel independence has been demonstrated, specific exceptions may be approved by the NRC staff to permit one of the two-out-of-four channels to be bypassed for an extended period of time.~~

Allowable Values for ESFAS Instrumentation (Analog) Functions are specified in Table 3.3.~~4~~-1. ~~[LTSPs]~~ and the methodologies for calculation of the as-left and as-found ~~tolerances~~ are described in ~~[insert the name of a document controlled under 10 CFR 50.59 such as~~ the Technical Requirements Manual~~or any document incorporated into the facility FSAR]~~. The ~~[LTSPs]~~ are selected to ensure that the actual setpoints remain conservative with respect to the as-found ~~tolerance~~ band between successive CHANNEL CALIBRATIONS. After each calibration the trip setpoint shall be left within the as-left band around the ~~[LTSP]~~.

The Bases for the LCO on ESFAS Functions are:

1. Safety Injection Actuation Signal

   a. Containment Pressure - High

      This LCO requires four channels of SIAS Containment Pressure - High to be OPERABLE in MODES 1, 2, and 3.

      The Allowable Value for this trip is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an offnormal condition. The setting is low enough to initiate the ESF Functions when an offnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

   b. Pressurizer Pressure - Low

      This LCO requires four channels of SIAS Pressurizer Pressure - Low to be OPERABLE in MODES 1, 2, and 3.

LCO (continued)

The Allowable Value for this trip is set low enough to prevent actuating the SIAS during normal plant operation and pressurizer pressure transients. The setting is high enough that with a LOCA or MSLB it will actuate to perform as expected, mitigating the consequences of the accidents.

The Pressurizer Pressure - Low trip may be blocked when pressurizer pressure is reduced during controlled plant shutdowns. This block is permitted below ~~1800~~ 1836 psia, and block permissive responses are annunciated in the control room. This allows for a controlled depressurization of the RCS, while maintaining administrative control of ESF protection. From a blocked condition, the block will be automatically removed as pressurizer pressure increases above ~~1800~~ 1836 psia, as sensed by two of the four sensor subsystems~~, in accordance with the bypass philosophy of removing bypasses when the enabling conditions are no longer satisfied~~. 1

This LCO requires four channels of the bypass permissive removal for SIAS Pressurizer Pressure - Low to be OPERABLE in MODES 1, 2, and 3.

The bypass permissive channels consist of four sensor subsystems and two actuation subsystems. This LCO applies to failures in the four sensor subsystems, including sensors, bistables, and associated equipment. Failures in the actuation subsystems, including the manual bypass key switches, are considered Actuation Logic failures and are addressed in LCO 3.3.~~5~~ 4. 3

This LCO applies to the bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue.

The block permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow blocking prior to reaching the trip setpoint.

2. Containment Spray Actuation Signal

CSAS is initiated either manually or automatically. ~~At many plants,~~ it is also necessary to have an automatic or manual SIAS for complete actuation. ~~The SIAS opens the containment spray valves, whereas the CSAS actuates other required components.~~ The SIAS requirement should always be satisfied on a legitimate CSAS, since 1 1

BASES

LCO (continued)

lower than the Containment Pressure – High-High signal

The parameters used for CSAS (Containment Pressure High-High and SIAS) give direct indication of a LOCA. The time to actuate CSAS depends on the break size. The Containment Pressure High-High set point is the controlling parameter since for all break sizes it takes longer to reach this set point than the SIAS set point.

the Containment Pressure - High signal setpoint used in the SIAS is ~~the same~~ setpoint used in the CSAS. ~~At many plants, the transmitters used to initiate CSAS are independent of those used in the SIAS to prevent inadvertent containment spray due to failures in two sensor channels.~~ 1

a. Containment Pressure - High -High 1

This LCO requires four channels of CSAS Containment Pressure - High to be OPERABLE in MODES 1, 2, and 3. -High 1

The Allowable Value is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an offnormal condition. The setting is low enough to initiate the ESF Functions when an offnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

~~The Containment Pressure - High setpoint is the same in the SIAS (Function 1), CSAS (Function 2), and CIAS (Function 3). However, different sensors and logic are used in each of these Functions.~~ 1

3. Containment Isolation Actuation Signal 1

a. Containment Pressure - High

This LCO requires four channels of CIAS Containment Pressure - High to be OPERABLE in MODES 1, 2, and 3. 1

The Allowable Value is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an offnormal condition. The setting is low enough to initiate the ESF Functions when an offnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

The Containment Pressure - High setpoint is the same in the SIAS (Function 1)~~, CSAS (Function 2),~~ and CIAS (Function 3). However, different ~~sensors and~~ logic ~~are~~ is used in each of these Functions. 1

LCO (continued)

b. Containment Radiation - High

This LCO requires four channels of CIAS Containment Radiation - High to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value is high enough to avoid unnecessary actuation, but adequate to provide diverse actuation of the CIAS in the event of a LOCA. R1

4. Main Steam Isolation Signal

The MSIS is required to be OPERABLE in MODES 1, 2, and 3 except when all associated valves are closed and de-activated.

a. Steam Generator Pressure - Low

This LCO requires four channels of MSIS Steam Generator Pressure - Low for each steam generator to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value is set below the full load operating value for steam pressure so as not to interfere with normal plant operation. However, the setting is high enough to provide the required protection for excessive steam demand. An excessive steam demand causes the RCS to cool down, resulting in a positive reactivity addition to the core. An MSIS is required to prevent the excessive cooldown.

This Function may be manually blocked when steam generator pressure is reduced during controlled plant cooldowns. The block is permitted below ~~785~~ psia, and block permissive responses are annunciated in the control room. This allows a controlled depressurization of the secondary system, while maintaining administrative control of ESF protection. From a blocked condition, the block will be removed automatically as steam generator pressure increases above ~~785~~ psia, as sensed by two of the four sensor subsystems~~, in accordance with the bypass philosophy of removing bypasses when the enabling conditions are no longer satisfied~~.

700 700 1 1 1

This LCO requires four channels per steam generator of the bypass removal for MSIS Steam Generator Pressure - Low to be OPERABLE in MODES 1, 2, and 3.

LCO (continued)

The bypass removal channels consist of four sensor subsystems and two actuation subsystems. This LCO applies to failures in the four sensor subsystems, including sensors, bistables, and associated equipment. Failures in the actuation subsystems, including the manual bypass key switches, are considered Actuation Logic failures and are addressed in LCO 3.3.~~5~~. 4 3

This LCO applies to the bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue.

The block permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow blocking prior to reaching the trip setpoint.

INSERT 1A 1

5. Recirculation Actuation Signal

a. Refueling Water Tank Level - Low

This LCO requires four channels of RWT Level - Low to be OPERABLE in MODES 1, 2, and 3.

The upper limit on the Allowable Value for this trip is set low enough to ensure RAS does not initiate before sufficient water is transferred to the containment sump. Premature recirculation could impair the reactivity control Function of safety injection by limiting the amount of boron injection. Premature recirculation could also damage or disable the recirculation system if recirculation begins before the sump has enough water to prevent air ~~containment~~ entrainment in the suction. The lower limit on the RWT Level - Low trip Allowable Value is high enough to transfer suction to the containment sump prior to emptying the RWT. 1

6. Auxiliary Feedwater Actuation Signal

The AFAS logic actuates auxiliary feedwater (AFW) to a steam generator on low level in that generator unless it has been identified as being ~~ruptured~~ faulted.

A low level in ~~either~~ the faulted steam generator, as sensed by each steam generator, a two-out-of-four coincidence of four wide range sensors for ~~any~~ generator, will generate an AFAS start signal, which starts ~~both~~ the affected trains of AFW 1 1

1

INSERT 1A

b. Containment Pressure - High

This LCO requires four channels of MSIS Containment Pressure - High to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value is set high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) and is not indicative of an offnormal condition. The setting is low enough to initiate the ESF Functions when an offnormal condition is indicated. This allows the ESF systems to perform as expected in the accident analyses to mitigate the consequences of the analyzed accidents.

BASES

LCO (continued)

the affected

, and the feedwater header differential pressure in both feedwater headers,

pumps and feeds ~~both~~ steam generator~~s~~. The AFAS also monitors the secondary differential pressure in both steam generators and initiates an AFAS block signal to a ~~ruptured~~ generator, if the pressure in ~~that generator is lower than that in the other generator~~ by the differential pressure setpoint.

one steam generator is lower than the other steam generator, or the pressure in one feedwater header is lower than the other feedwater header,

faulted

a. Steam Generator A/B Level - Low

This LCO requires four channels for each steam generator of Steam Generator Level - Low to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value ensures adequate time exists to initiate AFW while the steam generators can function as a heat sink.

b. Steam Generator Pressure Difference - High (SG-A > SG-B) or (SG-B > SG-A)

This LCO requires four channels per steam generator of Steam Generator Pressure Difference - High to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value for this trip is high enough to allow for small pressure differences and normal instrumentation errors between the steam generator channels during normal operation without an actuation. The setting is low enough to detect and inhibit feeding of a ~~ruptured~~ steam generator in the event of an MSLB or FWLB, while permitting the feeding of the intact steam generator.

faulted

INSERT 1

The ESFAS channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

APPLICABILITY All ESFAS Functions are required to be OPERABLE in MODES 1, 2, and 3. In MODES 1, 2, and 3 there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the main steam isolation valves to preclude a positive reactivity addition,
- Actuate AFW to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available),
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or MSLB, and

1

INSERT 1

b. Feedwater Header Pressure Difference - High
(FW-A > FW-B) or (FW-B > FW-A)

This LCO requires four channels per feedwater subsystem of Feedwater Header Pressure Difference - High to be OPERABLE in MODES 1, 2, and 3.

The Allowable Value for this trip is high enough to allow for small pressure differences and normal instrumentation errors between the feedwater header channels during normal operation without an actuation. The setting is low enough to detect and inhibit feeding of a faulted steam generator in the event of an MSLB or FWLB, while permitting the feeding of the intact steam generator.

BASES

APPLICABILITY (continued)

- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

In MODES 4, 5, and 6, automatic actuation of ESFAS Functions is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating the ESF components, if required, as addressed by LCO 3.3.~~5~~. In LCO 3.3.~~5~~, manual capability is required for Functions other than AFAS in MODE 4, even though automatic actuation is not required. Because of the large number of components actuated on each ESFAS, actuation is simplified by the use of the Manual ~~Trip~~ push buttons. Manual ~~Trip~~ of AFAS is not required in MODE 4 because AFW or shutdown cooling will already be in operation in this MODE.

The ESFAS Actuation Logic must be OPERABLE in the same MODES as the automatic and Manual ~~Trip~~. In MODE 4, only the portion of the ESFAS logic responsible for the required Manual ~~Trip~~ must be OPERABLE.

In MODES 5 and 6, ESFAS initiated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis.

Typically, the drift is small and results in a delay of actuation rather than a total loss of function. Determination of setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the actual trip setpoint is nonconservative with respect to the Allowable Value in Table 3.3.~~4~~-1, the channel is inoperable and the appropriate Condition(s) are entered.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value in Table 3.3.~~4~~-1, or the channel is not functioning as required, or the sensor, instrument loop, signal processing electronics, or ESFAS bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the plant must enter the Condition statement for the particular protection Function affected.

BASES

ACTIONS (continued)

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A Note has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.~~4~~-1. Completion Times for the inoperable channel of a Function will be tracked separately.

INSERT A

~~[A.1~~

INSERT B

~~Condition A applies to one CSAS Containment Pressure - High channel inoperable. CSAS logic is identical to that of the other ESFAS Functions; however, the inadvertent actuation of a CSAS is undesirable, since it may damage equipment inside containment. For this reason, placing the inoperable channel in trip is not an option as it is in Conditions B and C.]~~

INSERT C

~~[For those plants in which the SIAS is required for a complete CSAS actuation, Condition B for one ESFAS channel inoperable and Condition C for two ESFAS channels inoperable may be preferable to Condition A.~~

~~If one CSAS channel is inoperable, operation is allowed to continue, providing the inoperable channel is placed in bypass within 1 hour. The Completion Time of 1 hour allotted to bypass the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.]~~

~~B~~.1, ~~B~~.2.1, and ~~B~~.2.2

Condition ~~B~~ applies to the failure of a single channel of one or more input parameters in the following ESFAS Functions:

1. Safety Injection Actuation Signal
   Containment Pressure - High
   Pressurizer Pressure – Low
2. Containment Spray Actuation Signal
   Containment Pressure - High-High
3. Containment Isolation Actuation Signal
   Containment Pressure - High
   Containment Radiation - High

1

INSERT A

A.1, A.2.1, and A.2.2

Condition A applies to the failure of a single channel of one or more input parameters in the Containment Spray Actuation Signal (CSAS) ESFAS Function. R1

CSAS coincidence logic is normally two-out-of-four. If one CSAS channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one CSAS channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action A.1). With one channel in bypass, no additional random failure of a single channel could spuriously cause an actuation and a valid trip signal can still cause an actuation. With one channel in trip, an additional random failure of a single channel could spuriously cause an actuation. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to bypass or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

One failed channel is restored to OPERABLE status or is placed in trip within 48 hours (Required Action A.2.1 or A.2.2). Required Action A.2.1 restores the full capability of the function. Required Action A.2.2 places the function in a one-out-of-three configuration. In this configuration, common cause failure of the dependent channel cannot prevent ESFAS actuation. The 48 hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 48 hour period is a low probability event.

1

INSERT B

B.1 and B.2

Condition B applies to the failure of a single channel of one or more input parameters in the Containment Sump Recirculation Actuation Signal (RAS) ESFAS Function.

ESFAS coincidence logic is normally two-out-of-four. If one ESFAS channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one ESFAS channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action B.1). With one channel in bypass, no additional random failure of a single channel could spuriously cause an actuation and a valid trip signal can still cause an actuation. With one channel in trip, an additional random failure of a single channel could spuriously cause an actuation. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to bypass or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

The failed channel is restored to OPERABLE status within 48 hours (Required Action B.2). Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program (Required Action B.2). The 48 hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 48 hour period is a low probability event.

1

INSERT C

C.1, C.2.1, C.2.2.1, C.2.2.2

Condition C applies to the failure of a single channel of one or more input parameters in the Auxiliary Feedwater Actuation Signal (AFAS) ESFAS Function.

AFAS coincidence logic is normally two-out-of-four. If one AFAS channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one AFAS channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action C.1). With one channel in bypass, no additional random failure of a single channel could spuriously cause an actuation and a valid trip signal can still cause an actuation. With one channel in trip, an additional random failure of a single channel could spuriously cause an actuation. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to bypass or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

One failed channel is restored to OPERABLE status or is placed in bypass within 48 hours (Required Action C.2.1 or C.2.2.1). Required Action C.2.1 restores the full capability of the function. Required Action C.2.2.1 places the function in a two-out-of-three configuration. In this configuration, common cause failure of the dependent channel cannot prevent AFAS actuation. The 48 hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 48 hour period is a low probability event. R1

Additionally, the failed channel must be restored to OPERABLE status prior to entering MODE 3 following the next MODE 5 entry. The Completion Time of Required Action C.2.2.2 is based on adequate channel to channel independence, which allows a two-out-of-three channel operation since no single failure will cause or prevent a reactor trip. However, it is expected that the inoperable channel will be restored to OPERABLE status at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying restoration of the channel as well as any plant configuration changes required or shutting the plant down to repair the channel) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to repair the channel. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and NRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

Condition C is modified by a Note requiring Required Action C.2.2.2 to be completed whenever the Condition is entered even when the LCO is no longer applicable to ensure the channel is restored to OPERABLE prior to the next reactor startup from MODE 5 conditions.

BASES

ACTIONS (continued)

4. Main Steam Isolation Signal
   Steam Generator Pressure - Low

5. ~~Recirculation Actuation Signal~~
   ~~Refueling Water Tank Level - Low~~

6. ~~Auxiliary Feedwater Actuation Signal~~
   ~~Steam Generator Level - Low~~
   ~~Steam Generator Pressure Difference - High~~

ESFAS coincidence logic is normally two-out-of-four. If one ESFAS channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one ESFAS channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel is placed in bypass or trip within 1 hour (Required Action ~~B~~.1). With one channel in bypass, no additional random failure of a single channel could spuriously ~~trip the reactor~~ cause an actuation and a valid trip signal can still ~~trip the reactor~~. With one channel in trip, an additional random failure of a single channel could spuriously ~~trip the reactor~~ cause an actuation. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to bypass or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

INSERT D

~~One failed channel is restored to OPERABLE status or is placed in trip within [48] hours [or in accordance with the Risk Informed Completion Time Program] (Required Action B.2.1 or B.2.2). Required Action B.2.1 restores the full capability of the function. Required Action B.2.2 places the function in a one-out-of three configuration. In this configuration, common cause failure of the dependent channel cannot prevent ESFAS actuation. The [48] hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the [48] hour period is a low probability event. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~

INSERT E

~~C~~.1 and ~~C~~.2

Condition ~~C~~ applies to the failure of two channels in any of the following ESFAS functions:

1

INSERT D

Additionally, the failed channel must be restored to OPERABLE status prior to entering MODE 3 following the next MODE 5 entry. The Completion Time of Required Action D.2 is based on adequate channel to channel independence, which allows a two-out-of-three channel operation since no single failure will cause or prevent a reactor trip. However, it is expected that the inoperable channel will be restored to OPERABLE status at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying restoration of the channel as well as any plant configuration changes required or shutting the plant down to repair the channel) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to repair the channel. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and NRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

Condition D is modified by a Note requiring Required Action D.2 to be completed whenever the Condition is entered even when the LCO is no longer applicable to ensure the channel is restored to OPERABLE prior to the next reactor startup from MODE 5 conditions.

1

INSERT E

E.1 and E.2

Condition E applies to failure of two or more channels in the Auxiliary Feedwater Actuation Signal (AFAS) ESFAS Function or two channel in the Containment Spray Actuation System (CSAS) ESAFS Function.

With two inoperable channels, one channel should be placed in bypass, and the other channel should be placed in trip within the 1 hour Completion Time. With one channel of protective instrumentation bypassed, the ESFAS Function is in two-out-of-three logic, but with another channel failed the ESFAS may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESFAS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, ESFAS actuation will occur.

One of the failed channels should be restored to OPERABLE status within 48 hours. The 48 hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 48 hour period is a low probability event. After one channel is restored to OPERABLE status, the provisions of Condition C still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action E.2 must be placed in bypass if more than 48 hours has elapsed since the initial channel failure. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

BASES

ACTIONS (continued)

1. Safety Injection Actuation Signal
   Containment Pressure - High
   Pressurizer Pressure - Low

3. Containment Isolation Actuation Signal
   Containment Pressure - High
   Containment Radiation - High

4. Main Steam Isolation Signal
   Steam Generator Pressure - Low
   Containment Pressure - High
5. Recirculation Actuation Signal
   Refueling Water Tank Level - Low

~~6. Auxiliary Feedwater Actuation Signal~~
~~Steam Generator Level - Low~~
~~Steam Generator Pressure Difference - High~~

With two inoperable channels, one channel should be placed in bypass, and the other channel should be placed in trip within the 1 hour Completion Time. With one channel of protective instrumentation bypassed, the ESFAS Function is in two-out-of-three logic, but with another channel failed the ESFAS may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESFAS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, ESFAS actuation will occur.

One of the failed channels should be restored to OPERABLE status within ~~[~~48~~]~~ hours, for reasons similar to those stated under Condition ~~B~~. After one channel is restored to OPERABLE status, the provisions of Condition ~~B~~ still apply to the remaining inoperable channel. Therefore, except for Function 5 - RAS the channel that is still inoperable after completion of Required Action ~~C~~.2 must be placed in trip if more than ~~[~~48~~]~~ hours has elapsed since the initial channel failure. Function 1 - SIAS, Function 3 – CIAS, Function 4 – MSIS

After one Function 5 – RAS channel is restored to OPERABLE status, the provisions of Condition B still apply to the remaining inoperable channel. The RAS channel that is still inoperable after completion of Required Action F.2 must be restored to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time since the initial channel failure.

~~D~~.1, ~~D~~.2.1, ~~D~~.2.2.1, and ~~D~~.2.2.2

Condition ~~D~~ applies to the failure of one bypass removal channel.

The bypass removal channels consist of four sensor subsystems and two actuation subsystems. Condition ~~D~~ applies to failures in one of the four sensor subsystems, including sensors, bistables, and associated

BASES

ACTIONS (continued)

equipment. Failures in the actuation subsystems, including the manual bypass key switches, are considered Actuation Logic failures and are addressed in LCO 3.3.~~5~~.

In Condition ~~D~~, it is permissible to continue operation with one bypass permissive removal channel failed, providing the bypass is disabled (Required Action ~~D~~.1). This can be accomplished by removing the bypass with the manual bypass key switch, which disables the bypass in its respective ~~both~~ train~~s~~. Since the bypass Function must be manually enabled, the bypass permissive Function will not by itself cause an undesired bypass insertion.

Alternatively, the bypass may be disabled by defeating the bypass permissive input in one of the four channels to the two-out-of-four bypass removal logic, placing the bypass removal feature in one-out-of-three logic. Thus, any of the remaining three channels is capable of removing the bypass feature when the bypass enable conditions are no longer valid.

If the bypass removal feature in the inoperable channel cannot be defeated, actions to address the inoperability of the affected automatic trip channel must be taken. Required Action ~~D~~.2.1, Required Action ~~D~~.2.2.1, and Required Action ~~D~~.2.2.2 are equivalent to the Required Actions for a single automatic trip channel failure (Condition ~~B~~). The Completion Times have the same bases as discussed for Condition ~~B~~. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~

~~E~~.1, ~~E~~.2.1, and ~~E~~.2.2

Condition ~~E~~ applies to two inoperable bypass removal channels. The bypass removal channels consist of four sensor subsystems and two actuation subsystems. This Condition applies to failures in two of the four sensor subsystems. With two of the four sensor subsystems failed in a nonconservative direction (enabling the bypass Function), the bypass removal feature is in two-out-of-two logic. Failures in the actuation subsystems, including the manual bypass key switches, are considered Actuation Logic failures and are addressed in LCO 3.3.~~5~~.

In Condition ~~E~~, it is permissible to continue operation with two bypass permissive channels failed, providing the bypasses are disabled in a similar manner as discussed for Condition ~~D~~.

BASES

ACTIONS (continued)

If the failed bypasses cannot be disabled, actions to address the inoperability of the affected automatic trip channels must be taken. Required Action ~~E~~.2.1 and Required Action ~~E~~.2.2 are equivalent to the Required Actions for a two automatic trip channel failure (Condition ~~C~~). Also similar to Condition ~~C~~, after one set of inoperable channels is restored, the provisions of Condition ~~D~~ still apply to the remaining inoperable channel, with the Completion Time measured from the point of the initial bypass channel failure. The Completion Times have the same bases as discussed for Condition ~~C~~. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~

~~F~~.1 and ~~F~~.2

If the Required Actions and associated Completion Times of Condition A, B, C, D, ~~or~~ E are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within ~~[~~12~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for any particular ESFAS Function are found in the SRs column of Table 3.3.~~4~~-1 for that Function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and response time testing.

~~REVIEWER'S NOTE~~

~~In order for a unit to take credit for topical reports as the basis for justifying Frequencies, topical reports should be supported by an NRC staff Safety Evaluation Report that establishes the acceptability of each topical report for that unit.~~

~~REVIEWER'S NOTE~~

~~Notes a and b are applied to the setpoint verification Surveillances for each ESFAS Instrumentation (Analog) Function in Table 3.3.4-1 unless one or more of the following exclusions apply:~~

~~1. Manual actuation circuits, automatic actuation logic circuits or instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.~~

~~2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.~~ 4

~~3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.~~

3

SR 3.3.~~4~~.1 3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

~~[The Frequency of about once every shift is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~CHANNEL CHECK supplements less formal, but more frequent, checks of CHANNEL OPERABILITY during normal operational use of displays associated with the LCO required channels.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

3

SR 3.3.~~4~~.2 3

~~[~~ ~~A CHANNEL FUNCTIONAL TEST is performed every [92] days to ensure the entire channel will perform its intended function when needed.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

Move to end of this SR 3.3.3.2 description

A CHANNEL FUNCTIONAL TEST on each ESFAS instrument channel is performed to ensure the entire channel will perform its intended function when needed.

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. 1

The CHANNEL FUNCTIONAL TEST tests the individual sensor subsystems using an analog test input to each bistable.

BASES

SURVEILLANCE REQUIREMENTS (continued)

A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified ~~tolerance~~ around the setpoint. Any setpoint adjustment shall be consistent with the assumptions of the plant specific setpoint analysis.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. ~~The requirements for this review are outlined in Reference [10].~~

SR 3.3.~~4~~.2 is modified by two Notes as identified in Table 3.3.~~4~~-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ of the ~~[~~LTSP~~]~~. Where a setpoint more conservative than the ~~[~~LTSP~~]~~ is used in the plant surveillance procedures ~~[~~NTSP~~]~~, the as-left and as-found ~~tolerances~~, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ of the ~~[~~LTSP~~]~~, then the channel shall be declared inoperable.

The second Note also requires that ~~[~~LTSP~~]~~ and the methodologies for calculating the as-left and the as-found ~~tolerances~~ be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~.

SR 3.3.~~4~~.3

SR 3.3.~~4~~.3 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.~~4~~.2, except 3.3.~~4~~.3 is ~~performed within 92 days prior to startup and is~~ only applicable to bypass Functions. These include the Pressurizer Pressure - Low bypass and the MSIS Steam Generator Pressure - Low bypass. A successful test of the required contact(s) of a channel relay may be

SURVEILLANCE REQUIREMENTS (continued)

performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

~~The CHANNEL FUNCTIONAL TEST for proper operation of the bypass removal Functions is critical during plant heatups because the bypasses may be in place prior to entering MODE 3 but must be removed at the appropriate points during plant startup to enable the ESFAS Function. Consequently, just prior to startup is the appropriate time to verify bypass removal Function OPERABILITY. Once the bypasses are removed, the bypasses must not fail in such a way that the associated ESFAS Function is inappropriately bypassed. This feature is verified by the appropriate ESFAS Function CHANNEL FUNCTIONAL TEST.~~ 1

~~The allowance to conduct this Surveillance within 92 days of startup is based upon the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 11).~~ 1

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 1

3

SR 3.3.~~4~~.4 3

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the extension analysis. ~~The requirements for this review are outlined in Reference [10].~~ 1

~~[The Frequency is based upon the assumption of an [18] month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Move to end of this SR 3.3.3.4 description

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.3.~~4~~ 3.4 is modified by two Notes as identified in Table 3.3.~~4~~ 3-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found ~~tolerance~~ acceptance criteria band (e.g., outside the conservative side of the as-found acceptance criteria band). but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be returned to within the as-left ~~tolerance~~ acceptance criteria band of the ~~[LTSP]~~ NTSP. Where a setpoint more conservative than the ~~[LTSP]~~ NTSP is used in the plant surveillance procedures ~~[NTSP]~~ NTSP, the as-left and as-found ~~tolerances~~ acceptance criteria bands, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and~~/or~~ Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left ~~tolerance~~ acceptance criteria band of the ~~[LTSP]~~ NTSP, then the channel shall be declared inoperable.

The second Note also requires that ~~[LTSP]~~ NTSP and the methodologies for calculating the as-left and the as-found ~~tolerances~~ acceptance criteria bands be in ~~[insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference]~~ the Technical Requirements Manual and Section 7.2 of the Updated Final Safety Analysis Report, respectively.

SR 3.3.~~4~~ 3.5

This Surveillance ensures that the train actuation response times are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. The analysis models the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment in both trains reaches the required functional state (e.g.,

SURVEILLANCE REQUIREMENTS (continued)

pumps at rated discharge pressure, valves in full open or closed position). Response time testing acceptance criteria are included in Reference 5. The test may be performed in one measurement or in overlapping segments, with verification that all components are measured.

~~---------- REVIEWER'S NOTE ----------~~
~~Applicable portions of the following TS Bases are applicable to plants adopting CEOG Topical Report CE NPSD-1167-1, "Elimination of Pressure Sensor Response Time Testing Requirements," and the methodology contained in Attachment 1 to TSTF-569.~~
~~----------~~ 2

Response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. ~~Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements," (Ref. 12) provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the Topical Report. The response time may be verified for components that replace the components that were previously evaluated in Ref. 12 provided that the components have been evaluated in accordance with the NRC approved methodology as discussed in Attachment 1 to TSTF-569, "Methodology to Eliminate Pressure Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing," (Ref. 13). Response time verification for other sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and reverified after maintenance that may adversely affect the sensor response time.~~ 1

~~[ESF RESPONSE TIME tests are conducted on a STAGGERED TEST BASIS of once every [18] months. This results in the interval between successive tests of a given channel of n x 18 months, where n is the number of channels in the Function. Surveillance of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The [18] month STAGGERED TEST BASIS Frequency is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~ 2

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 2

1. Technical Requirements Manual

REFERENCES 2

~~1~~. Regulatory Guide 1,105, "Setpoints for Safety-Related Instrumentation," Revision 3.

3 ~~2~~. 10 CFR 50, Appendix A.

4 ~~3~~. 10 CFR 100. 50.67

5 ~~4~~ U FSAR, Section ~~[~~7.3~~]~~.

~~5. NRC Safety Evaluation Report, [Date].~~

6. IEEE Standard 279-1971.

7 U FSAR, Chapter ~~[14]~~. 15

8. 10 CFR 50.49.

9. ~~"Plant Protection~~ IC-3.17, FPL Setpoint Standard. ~~System Selection of Trip Setpoint Values."~~

~~10. FSAR, Section [7.2].~~

~~11. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989.~~

~~12. CEOG Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements."~~

~~13.~~ Attachment 1 to TSTF-569, "Methodology to Eliminate Pressure ~~Sensor and Protection Channel (for Westinghouse Plants only) Response Time Testing."~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.3, BASES, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The heading for ISTS 3.3.4 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2). Therefore, ISTS 3.3.2 is not included in the PSL ITS. The successive Specifications are renumbered as necessary.

4. PSL does not use the terms “as-found tolerance” and “Limiting Trip Setpoint (LTSP)” in plant specific instrument calculations or the Updated Final Safety Analysis Report. Therefore, the use of these terms in Footnotes (a) and (b) in ISTS Table 3.3.1-1 are revised in the ITS to include the plant specific terminology. To comply with the guidance provided in NRC Regulatory Issue Summary 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36, ‘Technical Specifications,’ Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels,” the PSL setpoint calculations have been structured to include a Nominal Trip Setpoint (NTSP), also known as the field trip setpoint, and determination of an OPERABILITY limit range, referred to in CTS as acceptance criteria band. For PSL, the Nominal Trip Setpoint (NTSP) is synonymous with the Limiting Trip Setpoint (LTSP) and the as-found acceptance criteria band (i.e., OPERABILITY limit range) is synonymous with the as-found tolerance.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.3, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 4

3.3.4, ESFAS LOGIC AND MANUAL ACTUATION

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.4
Table 3.3.4-1

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

LCO 3.3.4
Table 3.3.4-1

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

Table 3.3.4-1

~~a.~~ With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, ~~declare the channel inoperable and~~ apply the applicable ACTION requirement of Table 3.3-3 ~~until the channel is restored to OPERABLE status~~ with the trip setpoint adjusted consistent with the Trip Setpoint value.

ACTIONS for associated Table 3.3.4-1 Functions 1-6

ACTIONS for associated Table 3.3.4-1 Functions 1-6

~~b.~~ With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1
SR 3.3.4.2

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes ~~and at the frequencies~~ shown in Table 4.3-2.

In accordance with the Surveillance Frequency Control Program

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

See ITS 3.3.3

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per function.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LA01

A02

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
ACTION E, F — a. Manual (Trip Buttons)	~~2~~	~~1~~	2	1, 2, 3, 4	8
b. Containment Pressure – High	4	2	3	1, 2, 3	9
c. Pressurizer Pressure – Low	4	2	3	1, 2, 3(a)	9
ACTION E, F — Add: Automatic Actuation Logic			2	1,2,3,4	ACTION E, F
2. CONTAINMENT SPRAY (CSAS)					
ACTION E, F — a. Manual (Trip Buttons)	~~2~~	~~1~~	2	1, 2, 3, 4	8
b. Containment Pressure – High-High	4	2(b)	3	1, 2, 3	10A, 10B
ACTION E, F — Add: Automatic Actuation Logic			2	1,2,3,4	ACTION E, F
3. CONTAINMENT ISOLATION (CIS)					
ACTION E, F — a. Manual (Trip Buttons)	~~2~~	~~1~~	2	1, 2, 3, 4	8
b. Containment Pressure – High	4	2	3	1, 2, 3	9
c. Containment Radiation – High	4	2	3	1, 2, 3, 4	9
d. SIAS	---------- (See Functional Unit 1 above) ----------				
ACTION E, F — Add: Automatic Actuation Logic			2	1,2,3,4	ACTION E, F
4. MAIN STEAM LINE ISOLATION (MSIS)					
ACTION C, D — a. Manual (Trip Buttons)	~~2/steam generator~~	~~1/steam generator~~	2~~/operating steam generator~~	1, 2, 3, ~~4~~	C, D → ~~8~~
b. Steam Generator Pressure – Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	9
ACTION C, D — Add: Automatic Actuation Logic			2	1,2[(a)],3[(a)]	ACTION C, D

(a) except when all MSIVs are closed

See ITS 3.3.3

L02 A03 L05

L01 A05

A01

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LA01

A02

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	~~2~~	~~1~~	2	1, 2, 3, 4	8
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	13
ACTION E, F ← Add: Automatic Actuation Logic			2	1,2,3,4	ACTION E, F
6. LOSS OF POWER					
a. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12
b. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12
c. 480 V Emergency Bus Under-voltage (Degraded Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12
7. AUXILIARY FEEDWATER (AFAS)					
ACTION A, B a. Manual (Trip Buttons)	~~4/SG~~	~~2/SG~~	~~4~~/SG	1, 2, 3	11
ACTION A, B b. Automatic Actuation Logic	~~4/SG~~	~~2/SG~~	2 ~~3~~/SG	1, 2, 3	11
c. SG Level (1A/1B) - Low	4/SG	2/SG	3/SG	1, 2, 3	14a, 14b, 15
8. AUXILIARY FEEDWATER ISOLATION					
a. SG 1A – SG 1B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	14a, 14b, 15
b. Feedwater Header 1A – 1B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	14a, 15

L02 A03 L05

See ITS 3.3.5

A05

See ITS 3.3.3

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1725 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1725 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed in this MODE below 685 psig; bypass shall be automatically removed at or above 685 psig.

See ITS 3.3.3

Add proposed LCO 3.3.4 ACTIONS Note — A06

Add proposed LCO 3.3.4 ACTIONS C and D — L02

ACTION STATEMENTS

One or more Functions with one Manual Actuation or Actuation Logic channel inoperable except AFAS and MSIS.

ACTION 8 - ~~With the number of OPERABLE channels one less than the Total Number of Channels,~~ restore ~~the inoperable~~ channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in~~ ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

ACTION E

ACTION F

MODE 5

MODE 3

36

Add proposed ACTION F Condition: One or more Functions with two Manual Actuation or Actuation Logic channels inoperable except AFAS and MSIS. — A04

ACTION 9 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

See ITS 3.3.3

A01

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION 10A - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

See ITS 3.3.3

a. The inoperable channel is placed in the bypassed or tripped condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then place the inoperable channel in the tripped condition.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

ACTION 10B - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

One or more Functions with one Auxiliary Feedwater Actuation Signal (AFAS) Manual Actuation or Actuation Logic channel inoperable.

ACTION 11 - ~~With the number of OPERABLE channels one less than the Total Number of Channels,~~ restore ~~the inoperable~~ channel~~s~~ to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours. ~~LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.~~

ACTION A

ACTION B

12

MODE 3

MODE 4

A07

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

See ITS 3.3.5

Add proposed ACTION B Condition: Two AFAS Manual Actuation or Actuation Logic channels inoperable.

A04

A01

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION 13 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

See ITS 3.3.3

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If OPERABILITY cannot be restored within 48 hours or in accordance with the Risk Informed Completion Time Program, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

ACTION 14 - With the number of channels OPERABLE one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If an inoperable SG level channel can not be restored to OPERABLE status within 48 hours, then AFAS-1 or AFAS-2 as applicable in the inoperable channel shall be placed in the bypassed condition. If an inoperable SG DP or FW Header DP channel can not be restored to OPERABLE status within 48 hours, then both AFAS-1 and AFAS-2 in the inoperable channel shall be placed in the bypassed condition. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

ACTION 15 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
1. SAFETY INJECTION (SIAS)			
~~a. Manual (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
b. Containment Pressure – High	≤ 5 psig	≤ 5 psig	See ITS 3.3.3
c. Pressurizer Pressure – Low	≥ 1600 psia	≥ 1600 psia	
2. CONTAINMENT SPRAY (CSAS)			
~~a. Manual (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
b. Containment Pressure – High-High	≤ 10 psig	≤ 10 psig	See ITS 3.3.3
3. CONTAINMENT ISOLATION (CIS)			
~~a. Manual (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
b. Containment Pressure – High	≤ 5 psig	≤ 5 psig	See ITS 3.3.3
c. Containment Radiation – High	≤ 10 R/hr	≤ 10 R/hr	
d. SIAS	----------(See FUNCTIONAL UNIT 1 above)----------		
4. MAIN STEAM LINE ISOLATION (MSIS)			
~~a. Manual (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
b. Steam Generator Pressure – Low	≥ 585 psig	≥ 585 psig	See ITS 3.3.3
5. CONTAINMENT SUMP RECIRCULATION (RAS)			
~~a. Manual RAS (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
b. Refueling Water Tank – Low	48 inches above tank bottom	48 inches above tank bottom	See ITS 3.3.3

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES	
6. LOSS OF POWER			
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	≥ 2900 volts with a 1 ± .5 second time delay	≥ 2900 volts with a 1 ± .5 second time delay	See ITS 3.3.5
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3831 volts with a 18 ± 2 second time delay	≥ 3831 volts with a 18 ± 2 second time delay	
c. 480 volts Emergency Bus Undervoltage (Degraded Voltage)	≥ 415 volts with a ≤ 9 second time delay	≥ 415 volts with a ≤ 9 second time delay	
7. AUXILIARY FEEDWATER (AFAS)			
~~a. Manual (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
~~b. Automatic Actuation Logic~~	~~Not Applicable~~	~~Not Applicable~~	
c. SG 1A & 1B Level Low	≥ 19.0%	≥ 18.0 %	
8. AUXILIARY FEEDWATER ISOLATION			See ITS 3.3.3
a. Steam Generator ΔP – High	≤ 275 psid	89.2 to 281 psid	
b. Feedwater Header High ΔP	≤ 150.0 psid	56.0 to 157.5 psid	

A01

~~DELETED~~

A01

~~DELETED~~

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	~~CHANNEL CHECK~~	~~CHANNEL CALIBRATION~~	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED	
	1. SAFETY INJECTION (SIAS)					
SR 3.4.4.2	a. Manual (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	~~N.A.~~	
	b. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
	c. Pressurizer Pressure – Low	SFCP	SFCP	SFCP	1, 2, 3	
SR 3.4.4.1	d. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1)	1, 2, 3	
	2. CONTAINMENT SPRAY (CSAS)					
SR 3.4.4.2	a. Manual (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	~~N.A.~~	
	b. Containment Pressure – -– High-High	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
SR 3.4.4.1	c. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1)	1, 2, 3	
	3. CONTAINMENT ISOLATION (CIS)					
SR 3.4.4.2	a. Manual (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	~~N.A.~~	
	b. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
	c. Containment Radiation – High	SFCP	SFCP	SFCP	1, 2, 3, 4	
SR 3.4.4.1	d. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1)	1, 2, 3	
	e. SIAS	N.A.	N.A.	SFCP	N.A.	See ITS 3.3.3
	4. MAIN STEAM LINE ISOLATION (MSIS)					
SR 3.4.4.2	a. Manual (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	~~N.A.~~	
	b. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
SR 3.4.4.1	c. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1)	1, 2, 3 (a) (a)	L01
	5. CONTAINMENT SUMP RECIRCULATION (RAS)					
SR 3.4.4.2	a. Manual RAS (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	~~N.A.~~	
	b. Refueling Water Storage Tank – Low	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
SR 3.4.4.1	c. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1)	1, 2, 3	

(a) except when all MSIVs are closed

L01

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	~~CHANNEL CHECK~~	~~CHANNEL CALIBRATION~~	FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED	
6. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	SFCP	SFCP	SFCP	1, 2, 3	
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3	
c. 480 V Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.5
7. AUXILIARY FEEDWATER (AFAS)					
SR 3.3.4.2 a. Manual (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	1, 2, 3	R1
b. SG Level (A/B) – Low	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
SR 3.3.4.1 c. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP	1, 2, 3	R1
8. AUXILIARY FEEDWATER ISOLATION					
a. SG Level (A/B) – Low and SG Differential Pressure (BtoA/AtoB) – High	N.A.	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
b. SG Level (A/B) – Low and Feedwater Header Differential Pressure (BtoA/AtoB) – High	N.A.	SFCP	SFCP	1, 2, 3	

A01

TABLE 4.3-2 (Continued)

TABLE NOTATION

~~(1) The logic circuits shall be tested manually in accordance with the Surveillance Frequency Control Program.~~

LA02

Add SR 3.3.4.1 Note 1

A08

Add SR 3.3.4.1 Note 2

L03

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.4
Table 3.3.4-1

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

LCO 3.3.4
Table 3.3.4-1

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

Table 3.3.4-1
ACTIONS for
Table 3.3.4-1 Functions 1-6

~~a.~~ With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, ~~declare the channel inoperable and~~ apply the applicable ACTION requirement of Table 3.3-3 ~~until the channel is restored to OPERABLE status~~ with the trip setpoint adjusted consistent with the Trip Setpoint value.

ACTIONS for
Table 3.3.4-1 Functions 1-6

~~b.~~ With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1
SR 3.3.4.2

~~4.3.2.1~~ Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES ~~and at the frequencies~~ shown in Table 4.3-2.

In accordance with the Surveillance Frequency Control Program

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillanc Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

See ITS 3.3.3

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per function.

A01

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LA01

A02

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
1. SAFETY INJECTION (SIAS)						
ACTION E, F — a. Manual (Trip Buttons)	~~2~~	~~1~~	2	1, 2, 3, 4	12	
b. Containment Pressure – High	4	2	3	1, 2, 3	13, 14	See ITS 3.3.3
c. Pressurizer Pressure – Low	4	2	3	1, 2, 3(a)	13, 14	
ACTION E, F — d. Automatic Actuation Logic	~~2~~	~~1~~	2	1, 2, 3, 4	12	
2. CONTAINMENT SPRAY (CSAS)						
ACTION E, F — a. Manual (Trip Buttons)	~~2~~	~~1~~	2	1, 2, 3, 4	12	
b. Containment Pressure – High-High	4	2	3	1(b), 2(b), 3(b)	18A, 18B	See ITS 3.3.3
ACTION E, F — c. Automatic Actuation Logic	~~2~~	~~1~~	2	1, 2, 3, 4	12	
3. CONTAINMENT ISOLATION (CIAS)						
ACTION E, F — a. Manual CIAS (Trip Buttons)	~~2~~	~~1~~	2	1, 2, 3, 4	12	
b. Safety Injection (SIAS)	See Functional Unit 1 for all Safety Injection Initiating Functions and Requirements					
c. Containment Pressure – High	4	2	3	1, 2, 3	13, 14	See ITS 3.3.3
d. Containment Radiation – High	4	2	3	1, 2, 3	13, 14	
ACTION E, F — e. Automatic Actuation Logic	~~2~~	~~1~~	2	1, 2, 3, 4	12	

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ CHANNELS OPERABLE	APPLICABLE MODES	ACTION
	4. MAIN STEAM LINE ISOLATION (MSIS)				(a) (a) L01	
ACTION C, D	a. Manual (Trip Buttons)	2	1	2	1, 2, 3	16
	b. Steam Generator Pressure – Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	13, 14
	c. Containment Pressure – High	4	2	3	1, 2, 3	13, 14
						See ITS 3.3.3
ACTION C, D	d. Automatic Actuation Logic	2	1	2	1, 2, 3	12
	5. CONTAINMENT SUMP RECIRCULATION (RAS)				(a) (a) L01	
ACTION E, F	a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12
	b. Refueling Water Tank - Low	4	2	3	1, 2, 3	19
						See ITS 3.3.3
ACTION E, F	c. Automatic Actuation Logic	2	1	2	1, 2, 3	12

(a) except when all MSIVs are closed — L01

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LA01

A02

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
6. LOSS OF POWER (LOV)						
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	17A	
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17B	See ITS 3.3.5
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17B	
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17B	
7. AUXILIARY FEEDWATER (AFAS)						
ACTION A, B a. Manual (Trip Buttons)	~~4~~/SG	~~2~~/SG	~~4~~/SG	1, 2, 3	15	
ACTION A, B b. Automatic Actuation Logic	~~4~~/SG	~~2~~/SG	~~3~~/SG	1, 2, 3	15	
c. SG Level (2A/2B) – Low	4/SG	2/SG	3/SG	1, 2, 3	20a, 20b, 21	
8. AUXILIARY FEEDWATER ISOLATION						
a. SG 2A – SG 2B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	20a, 20b, 21	See ITS 3.3.3
b. Feedwater Header 2A – 2B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	20a, 21	

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than 1836 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to 1836 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed in this MODE below 700 psia; bypass shall be automatically removed at or above 700 psia.

See ITS 3.3.3

Add proposed LCO 3.4.4 ACTIONS Note

A06

ACTION OF STATEMENTS

One or more Functions with one Manual Actuation or Actuation Logic channel inoperable except AFAS and MSIS.

ACTION E

ACTION F

ACTION 12 - ~~With the number of OPERABLE channels one less than the Total Number of Channels,~~ restore ~~the inoperable~~ channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in~~ ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

MODE 3

36

MODE 5

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

See ITS 3.3.3

Process Measurement Circuit	Functional Unit Bypassed
1. Containment Pressure -	Containment Pressure – High (SIAS, CIAS, CSAS) Containment Pressure – High (RPS)
2. Steam Generator Pressure -	Steam Generator Pressure – Low (MSIS) AFAS-1 and AFAS-2 (AFAS) Thermal Margin/Low Pressure (RPS) Steam Generator Pressure – Low (RPS)
3. Steam Generator Level -	Steam Generator Level – Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)
4. Pressurizer Pressure -	Pressurizer Pressure – High (RPS) Pressurizer Pressure – Low (SIAS) Thermal Margin/Low Pressure (RPS)

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

See ITS 3.3.3

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below.

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Containment Pressure -	Containment Pressure – High (SIAS, CIAS, CSAS) Containment Pressure – High (RPS)
2. Steam Generator Pressure -	Steam Generator Pressure – Low (MSIS) AFAS-1 and AFAS-2 (AFAS) Thermal Margin/Low Pressure (RPS) Steam Generator Pressure – Low (RPS)
3. Steam Generator Level -	Steam Generator Level – Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)
4. Pressurizer Pressure -	Pressurizer Pressure – High (RPS) Pressurizer Pressure – Low (SIAS) Th

One or more Functions with one Auxiliary Feedwater Actuation Signal (AFAS) Manual Actuation or Actuation Logic channel inoperable.

ACTION A

ACTION B

ACTION B.2 Note

12

MODE 3

MODE 4

A07

ACTION 15 - ~~With the number of OPERABLE channels one less than the Total Number of Channels,~~ restore ~~the inoperable~~ channel~~s~~ to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours. ~~LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.~~

ACTION C.1

ACTION 16 - ~~With the number of OPERABLE channels one less than the Total Number of Channels,~~ restore ~~the inoperable~~ channel to OPERABLE status within 48 hours or ~~declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5~~.

L03

A03

Add ACTION D

L04

in accordance with the Risk Informed Completion Time Program

One Main Steam Isolation Signal (MSIS) Manual Actuation or Actuation Logic channel inoperable.

ACTION 17A - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or place the inoperable channel in the tripped condition and verify that the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

See ITS 3.3.5

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 17B - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or place the inoperable channel in the tripped condition and verify that the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

See ITS 3.3.5

ACTION 18A - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then place the inoperable channel in the tripped condition.

b. With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.

See ITS 3.3.3

ACTION 18B - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 19 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. Within 1 hour the inoperable channel is placed in either the bypassed or tripped condition. If OPERABILITY cannot be restored within 48 hours or in accordance with the Risk Informed Completion Time Program, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

See ITS 3.3.3

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If an inoperable SG level channel can not be restored to OPERABLE status within 48 hours, then AFAS-1 or AFAS-2 as applicable in the inoperable channel shall be placed in the bypassed condition. If an inoperable SG DP or FW Header DP channel can not be restored to OPERABLE status within 48 hours, then both AFAS-1 and AFAS-2 in the inoperable channel shall be placed in the bypassed condition. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

b With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.

ACTION 21 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
1. SAFETY INJECTION (SIAS)			
~~a. Manual (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
b. Containment Pressure – High	≤ 3.5 psig	≤ 3.6 psig	See ITS 3.3.3
c. Pressurizer Pressure – Low	≥ 1736 psia	≥ 1728 psia	
~~d. Automatic Actuation Logic~~	~~Not Applicable~~	~~Not Applicable~~	
2. CONTAINMENT SPRAY (CSAS)			
~~a. Manual (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
b. Containment Pressure – High-High	≤ 5.40 psig	≤ 5.50 psig	See ITS 3.3.3
~~c. Automatic Actuation Logic~~	~~Not Applicable~~	~~Not Applicable~~	
3. CONTAINMENT ISOLATION (CIAS)			
~~a. Manual CIAS (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
b. Safety Injection (SIAS)	Not Applicable	Not Applicable	
c. Containment Pressure – High	≤ 3.5 psig	≤ 3.6 psig	See ITS 3.3.3
d. Containment Radiation – High	≤ 10 R/hr	≤ 10 R/hr	
~~e. Automatic Actuation Logic~~	~~Not Applicable~~	~~Not Applicable~~	
4. MAIN STEAM LINE ISOLATION			
~~a. Manual (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
b. Steam Generator Pressure – Low	≥ 600 psia	≥ 567 psia	See ITS 3.3.3
c. Containment Pressure – High	≤ 3.5 psig	≤ 3.6 psig	
~~d. Automatic Actuation Logic~~	~~Not Applicable~~	~~Not Applicable~~	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES	
5. CONTAINMENT SUMP RECIRCULATION (RAS)			
~~a. Manual RAS (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
b. Refueling Water Tank – Low	5.67 feet above tank bottom	4.62 feet to 6.24 feet above tank bottom	See ITS 3.3.3
~~c. Automatic Actuation Logic~~	~~Not Applicable~~	~~Not Applicable~~	
6. LOSS OF POWER			
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ 3120 volts	≥ 3120 volts	
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	≥ 360 volts	≥ 360 volts	See ITS 3.3.5
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	≥ 3848 volts with < 10-second time delay	≥ 3848 volts with < 10-second time delay	
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	≥ 432 volts	≥ 432 volts	
7. AUXILIARY FEEDWATER (AFAS)			
~~a. Manual (Trip Buttons)~~	~~Not Applicable~~	~~Not Applicable~~	
~~b. Automatic Actuation Logic~~	~~Not Applicable~~	~~Not Applicable~~	
c. SG 2A & 2B Level Low	≥ 19.0%	≥ 18.0 %	
8. AUXILIARY FEEDWATER ISOLATION			
a. Steam Generator ΔP – High	≤ 275 psid	89.2 to 281 psid	See ITS 3.3.3
b. Feedwater Header ΔP – High	≤ 150.0 psid	56.0 to 157.5 psid	

A01

~~DELETED~~

A01

~~DELETED~~

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~~DELETED~~

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	~~CHANNEL CHECK~~	~~CHANNEL CALIBRATION~~	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
SR 3.3.4.2	1. SAFETY INJECTION (SIAS)					
	a. Manual (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	1, 2, 3, 4	
	b. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
	c. Pressurizer Pressure – Low	SFCP	SFCP	SFCP	1, 2, 3	
SR 3.3.4.1	d. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1), SFCP(3)	1, 2, 3, 4	
	2. CONTAINMENT SPRAY (CSAS)					
SR 3.3.4.2	a. Manual (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	1, 2, 3, 4	
	b. Containment Pressure – High-High	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
SR 3.3.4.1	c. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1), SFCP(3)	1, 2, 3, 4	
	3. CONTAINMENT ISOLATION (CIAS)					
SR 3.3.4.2	a. Manual CIAS (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	1, 2, 3, 4	
	b. Safety Injection SIAS	N.A.	N.A.	SFCP	1, 2, 3, 4	
	c. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
	d. Containment Radiation – High	SFCP	SFCP	SFCP	1, 2, 3	
SR 3.3.4.1	e. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1), SFCP(3)	1, 2, 3, 4	
	4. MAIN STEAM LINE ISOLATION					L01
SR 3.3.4.2	a. Manual (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	1, 2, 3	
	b. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
	c. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3	
SR 3.3.4.1	d. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1), SFCP(3)	1, 2, 3~~, 4~~	
	5. CONTAINMENT SUMP RECIRCULATION (RAS)					
SR 3.3.4.2	a. Manual RAS (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	N.A.	
	b. Refueling Water Tank – Low	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
SR 3.3.4.1	c. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1), SFCP(3)	1, 2, 3	

(a) except when all MSIVs are closed

L01

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	~~CHANNEL CHECK~~	~~CHANNEL CALIBRATION~~	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
6. LOSS OF POWER (LOV)					
a. 4.16 kV and 480 V Emergency Bus Undervoltage (Loss of Voltage)	SFCP	SFCP	SFCP	1, 2, 3, 4	See ITS 3.3.3
b. 4.16 kV and 480 V Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3, 4	
7. AUXILIARY FEEDWATER (AFAS)					
SR 3.3.4.2 a. Manual (Trip Buttons)	~~N.A.~~	~~N.A.~~	SFCP	1, 2, 3	
b. SG Level (A/B) – Low	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
SR 3.3.4.1 c. Automatic Actuation Logic	~~N.A.~~	~~N.A.~~	SFCP(1), SFCP(2)	1, 2, 3	
8. AUXILIARY FEEDWATER ISOLATION					
a. SG Level (A/B) – Low and SG Differential Pressure (B to A/A to B) – High	N.A.	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
b. SG Level (A/B) – Low and Feedwater Header Differential Pressure (B to A/A to B) – High	N.A.	SFCP	SFCP	1, 2, 3	

TABLE NOTATION

SR 3.3.4.1 Note

~~(1)~~ Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay (solid-state component) and verification of the OPERABILITY of each initiation relay (solid-state component).

~~(2) An actuation relay test shall be performed which shall include the energization/de-energization of each actuation relay and verification of the OPERABILITY of each actuation relay.~~ LA02

~~(3) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Testing of the ESFAS subgroup relays shall be performed in accordance with the Surveillance Frequency Control Program.~~ LA02

Add SR 3.3.4.1 Note 2 L03

DISCUSSION OF CHANGES
ITS 3.3.4, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS)
LOGIC AND MANUAL ACTUATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.3.2.1 and Unit 2 CTS 3.3.2 specify requirements for ESFAS instrumentation and list the requirements for each functional unit in CTS Tables 3.3-3, 3.3-4, and 4.3-2, including the manual channels and automatic actuation logic (Unit 2 only). ITS 3.3.4 provides requirements for the ESFAS Actuation Logic and Manual Actuation channels and provides the requirements in the LCO, Applicability, ACTIONS, and Surveillance Requirements. Additionally, ITS Table 3.3.4-1 does not retain the "TOTAL NO. OF CHANNELS" or "CHANNELS TO TRIP" columns. DOC LA01 describes the change that moves the information of the "TOTAL NO. OF CHANNELS" and "CHANNELS TO TRIP" columns to the Bases. This changes the CTS by moving the "MINIMUM CHANNELS OPERABLE" requirements to the LCO statement and reformatting the content consistent with the ISTS. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 **Unit 1 only:** CTS Table 3.3-3 Functional Unit 1 (SIAS), Functional Unit 2 (CSAS), Functional Unit 3 (CIS), Functional Unit 4 (MSIS) and Functional Unit 5 (RAS) do not specifically include the Automatic Actuation Logic and Actuation Relay Function. However, CTS Table 4.3-2 list Surveillance Requirements for the ESFAS automatic actuation logic associated with these Functions. Therefore, per the requirements of CTS 4.0.1 (ITS SR 3.0.1), CTS 3.3.2.1 encompasses the ESFAS automatic actuation logic. ITS LCO 3.3.4 requires two ESFAS Actuation Logic channels to be OPERABLE in the MODES specified for each Function. This changes the CTS by explicitly requiring two channels of the ESFAS Actuation Logic Functions for SIAS, CSAS, CIS, MSIS, and RAS to be OPERABLE.

This change is considered acceptable because the SIAS, CSAS, CIS, MSIS and RAS Functions require the Actuation Logic channels to operate properly in order to actuate the associated function response. Two channels are required to be OPERABLE to ensure a single failure of a logic channel does not prevent the actuation of the associated SIAS, CSAS, CIS, MSIS or RAS function response. The proposed Applicability is consistent with the Applicability of the Functions listed under CTS Table 3.3-3 Functional Units 1.a, 2.a, 3.a, 5.a and 4.a as modified by DOC L02. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 Unit 1 CTS 3.3.2.1 does not currently contain an ACTION when two Manual Actuation channels are inoperable and Unit 2 CTS 3.3.2 does not currently

contain an ACTION when two Manual Actuation or Actuation Logic channels are inoperable; therefore, an LCO 3.0.3 entry would be required. ITS 3.3.4 ACTION B requires the unit to be in MODE 3 in 6 hours and MODE 4 in 12 hours when two Auxiliary Feedwater Actuation Signal (AFAS) Manual Actuation or Actuation Logic channels are inoperable. ITS 3.3.4 ACTION F requires the unit to be in MODE 3 in 6 hours and MODE 5 in 36 hours when either two Safety Injection Actuation Signal (SIAS), two Containment Spray actuation Signal (CSAS), two Containment Isolation Signal (CIS – Unit 1), two Containment Isolation Actuation Signal (CIAS – Unit 2), or two Containment Sump Recirculation Actuation Signal (RAS) Manual Actuation or Actuation Logic channels are inoperable. This changes the CTS by explicitly stating shutdown ACTIONS when two Manual Actuation or Actuation Logic channels are inoperable for an ESFAS Function.

The purpose of ITS 3.3.4 ACTIONS B, and F when two Manual Actuation or Actuation Logic channels are inoperable, is to ensure the plant is brought to a MODE in which the LCO for the ESFAS Functions does not apply within a reasonable amount of time in a controlled manner. Unit 1 CTS 3.3.2.1 and Unit 2 CTS 3.3.2 are silent on these actions, deferring to CTS 3.0.3 for the actions to accomplish this. The proposed change is acceptable because the ACTIONS specified in ITS 3.3.4 adopt ISTS structure for placing the unit outside the MODE of Applicability without changing the time specified to enter MODE 3, MODE 4, or MODE 5, as applicable. This change is designated as administrative because it does not result in a technical change to the CTS.

A05 **Unit 1 only:** CTS Table 3.3-3 Functional Unit 4.a provides requirements for MSIS manual actuation. The Total No. Of Channels and Minimum Channels Operable columns specify 2 channels per steam generator (SG) and the Channels to Trip column specifies 1 channel per SG. ITS LCO 3.3.4, in part, requires two channels of ESFAS Manual Actuation for each Function listed in Table 3.3.4-1 to be OPERABLE, which includes the main steam isolation signal (MSIS) Function. This changes the CTS by requiring two MSIS Manual Actuation channels instead of two channels per SG.

The purpose of the required number of channels is to ensure a manual MSIS trip can be initiated coincident with a single failure. The MSIS manual actuation function is comprised of one manual actuation push button per MSIS actuation logic. A single pushbutton associated with either Manual Actuation channel is required to initiate an isolation of both MSIVs. Each Manual Actuation channel actuates its associated MSIS Actuation Logic channel and each MSIS Actuation Logic channel provides a signal to both MSIVs via interposing relays. CTS Table 3.3-3 Functional Unit 4.a specifies each logic input to each MSIV as a Manual Actuation channel (i.e., two Manual Actuation channels to each SG). ISTS LCO 3.3.5 (ITS LCO 3.3.4) and the associated ACTIONS and Surveillances are formatted to address each individual manual actuation push button circuit to the MSIS logic channel as a channel. Actuating either MSIS Manual Actuation will result in a full main steam isolation. Therefore, changing the minimum number of required channels from 2 per steam generator to 2 results in an equivalent requirement; two manual push buttons must be OPERABLE and one manual push button is required to initiate a main steam isolation.

This change is a presentation preference is consistent with NUREG-1432, Rev. 5 and PSL Unit 2 CTS Table 3.3-1 Functional Unit 1 Total No. of Channels, Minimum Channels Operable, and Channels to Trip columns. This change is designated as administrative because it does not result in technical changes to the CTS.

A06 CTS Table 3.3-3 Actions describe the Actions to be taken when ESFAS Function manual actuations or actuation logic channels are inoperable. ITS 3.3.4 also describes Actions to be taken when ESFAS Function manual actuations or actuation logic channels are inoperable and contains a note that separate condition entry is allowed for each ESFAS Function. This changes the CTS by adding a Note stating that separate condition entry is allowed for each ESFAS Function, those Functions being the SIAS, CSAS, CIS (Unit 1) and CIAS (Unit 2), MSIS, RAS, and AFAS Functions.

The purpose of the CTS Actions is to provide the appropriate compensatory actions for inoperable ESFAS Functions. This proposed change will allow separate condition entry for each ESFAS Function. The Note clarifies that ESFAS Functions are treated as separate entities, each with separate Completion Times. This change is acceptable because it clearly states the current requirement. The CTS considers each ESFAS Function to be separate and independent from the others including on a per steam generator basis. This change is designated as administrative because it does not result in technical changes to the CTS.

A07 Unit 1 CTS Table 3.3-3 Action 11 and Unit 2 CTS Table 3.3-3 Action 15 describe the actions to be taken when one Manual Actuation or Actuation Logic channel associated with an AFAS Functional Unit cannot be restored to OPERABLE status within the required Completion Time. ACTION 15 includes a statement that “LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.” ITS 3.3.4 ACTION B also describes actions to be taken when one Manual Actuation or Actuation Logic channel associated with an AFAS Function cannot be restored to OPERABLE status within the required Completion Time but does not include the CTS statement regarding LCO 3.0.4.a because the AFAS instrumentation is not applicable in HOT SHUTDOWN (MODE 4). This changes the CTS by eliminating a note that is unnecessary for Technical Specification compliance.

The purpose of the CTS Actions is to provide the appropriate compensatory actions for inoperable AFAS Functions. This CTS statement that LCO 3.0.4.a is not appliable when entering Hot Shutdown is unnecessary. The purpose of CTS 3.0.4 (ITS LCO 3.0.4) is to prohibit entry into a MODE or other specified condition in which the LCO is applicable when the LCO is not met except as allowed by LCO 3.0.4.a, b, or c. CTS Table 3.3-3 Action 11 (Unit 1) and Action 15 (Unit 2) are only related to the AFAS Functions (CTS Table 3.3-3 Functional Units 7.a and 7.b). These Functions are not applicable in MODE 4 (Hot Shutdown). Therefore, LCO 3.0.4 does not apply to the AFAS Functions when in MODE 4 resulting in the statement related to LCO 3.0.4.a not being applicable meaningless. This change is designated as administrative because it removes a requirement that serves no purpose and does not result in technical changes to the CTS.

A08 **Unit 1 only:** CTS Table 4.3-2 requires CHANNEL FUNCTIONAL TEST requirements for each ESFAS Functional Unit, including ESFAS automatic actuation logic channels. ITS SR 3.3.4.1 also requires a CHANNEL FUNCTIONAL TEST on each ESFAS logic channel and includes a Note (Note 1) that states, "Testing of Actuation Logic shall include verification of the proper operation of each initiation relay." The added Note is consistent with the ISTS. This changes the CTS by explicitly stating that proper operation of each initiation relay is required to satisfy the CHANNEL FUNCTIONAL TEST requirement.

The purpose of the additional Note is to explicitly clarify the intent of the CHANNEL FUNCTIONAL TEST to include the initiation relays of the equipment. CTS 1.6 definition of a CHANNEL FUNCTIONAL TEST requires the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions. Similarly, the ITS definition of a CHANNEL FUNCTIONAL TEST states, in part, "…the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY." Although it is inferred by the definition of a CHANNEL FUNCTIONAL TEST that testing the associated initiation relays is required to verify logic OPERABILITY, Note 1 to SR 3.3.4.1 provides explicit clarification of the requirement. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-3 for ESFAS instrumentation has three columns stating various requirements for each function. These columns are labeled, "TOTAL NO. OF CHANNELS," "CHANNELS TO TRIP," and "MINIMUM CHANNELS OPERABLE." ITS Table 3.3.4-1 does not retain the "TOTAL NO. OF CHANNELS" or "CHANNELS TO TRIP" columns. This changes the CTS by moving the information of the "TOTAL NO. OF CHANNELS" and "CHANNELS TO TRIP" columns to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the number of required channels and the appropriate Condition to enter if a required

channel becomes inoperable. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS Table 4.3-2 contains Note (1) requiring, in part, that "logic circuits to be tested manually." Unit 2 CTS Table 4.3-2 contains Notes (2) and (3) modifying the Automatic Actuation Logic Channel Functional Test requirements. Note (2) requires an actuation relay test to be performed including the energization/de-energization of each actuation relay and verification of the OPERABILITY of each actuation relay. Note (3) requires a subgroup relay test to be performed including the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. ITS 3.3.4 does not retain these Notes, but rather, includes this testing procedural guidance in the ITS Bases. This changes the CTS by moving procedural details regarding testing to the Bases.

The purpose of the CTS Notes is to ensure the appropriate components are tested to demonstrate OPERABILITY of instrumentation actuation logic. The removal of these details, which are procedural details, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the CHANNEL FUNCTIONAL TEST requirement and the definition of a CHANNEL FUNCTIONAL TEST which ensures the appropriate components are tested to verify instrumentation OPERABILITY. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 (*Category 2 – Relaxation of Applicability*) Unit 1 CTS 3.3.2.1, Table 3.3-3 Functional Unit 4 – main steam isolation signal (MSIS) manual actuation is applicable in MODES 1, 2, 3, and 4 and Table 4.3-2 Functional Unit 4 – MSIS automatic actuation logic is applicable in MODES 1, 2, and 3. Unit 2 CTS 3.3.2, Tables 3.3-3 and 4.3-2 Functional Unit 4 –MSIS manual and automatic actuation logic are applicable in MODES 1, 2, and 3. When one MSIS manual actuation channel (or one automatic actuation logic channel for Unit 2) is inoperable, CTS Table 3.3-3 Action 8 (Unit 1) and Action 12 (Unit 2) require, in part, to restore the channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program.

ITS LCO 3.3.4 (Table 3.3.4-1) requires the MSIS manual actuation and automatic actuation channel to be OPERABLE in MODES 1, 2, and 3 except in MODES 2 and 3 when valves isolated by the MSIS Function are closed as specified in Footnote (a). Consistent with CTS, ITS 3.3.4 ACTION C requires restoring a channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program when one MSIS Manual Actuation or Actuation Logic channel is inoperable. However, ITS 3.3.4 ACTION D requires closing the main steam isolation valves (MSIVs) and the main feedwater isolation valves (MFIVs) in 6 hours, or be in MODE 3 in 6 hours and MODE 4 in 12 hours when two MSIS Manual Actuation or Actuation Logic channels are inoperable or when the action to restore one MSIS Manual Actuation or Actuation Logic channel cannot be completed within the required Completion Time. This changes the CTS by deleting the MODE 4 applicability requirement for Unit 1 MSIS, and changing the MODE 2 and 3 Applicability for Unit 1 and Unit 2 to include “except when valves isolated by the MSIS Function are closed.” This change aligns the Applicability of the MSIS actuation instrumentation with the applicability requirements of the associated MSIS instrument channels (i.e., steam generator low pressure instrument channels and the Unit 2 containment high pressure instrument channels) and the supported equipment (i.e., the MSIVs and MFIVs). This change also revises the CTS Actions to add appropriate actions to reflect the change to the Applicability, which provides sufficient remedial actions to safely continue operation in lieu of a plant cooldown.

The purpose of the MSIS manual and automatic instrumentation in CTS Table 3.3-3 is to ensure the ESFAS MSIS Functions are OPERABLE when the supported equipment is required to be OPERABLE. Automatic steam line and feedwater line isolation is assumed in the mitigation of a major secondary system pipe rupture accident (e.g., main steam line break or feedwater line break). The MSIS manual actuation instrumentation is not credited in the safety analysis and a safety analysis limit is not specified for this actuation function. MSIS automatic actuation instrumentation is credited in the safety analysis.

In MODES 1, 2, and 3, there is sufficient energy in the primary and secondary systems to warrant automatic ESF System response to ensure the main steam and feedwater lines can be isolated in the event of a high energy secondary system pipe rupture. The exception to MODE 2 and 3 is added to clarify that the MSIS manual and automatic actuation instrumentation is not required to be OPERABLE when the valves actuated by the MSIS instrumentation are in positions that support the safety analyses. When the valves isolated by the MSIS function are in the closed position, they are in their assumed accident position. This change is acceptable, because when the MSIVs and MFIVs are closed, the adverse effects of a high energy secondary system pipe rupture are precluded and the requirements continue to ensure that the structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses. The proposed applicability and actions provide the same level of protection as the current requirements for the supported equipment.

Since the proposed actions provide the allowance to place the MSIVs and MFIVs in their accident position with the option to continue a plant cooldown, sufficient remedial measures continue to be provided to allow safe operation pursuant the requirements of 10 CFR 50.36(c)(2). Proposed ITS 3.3.4 ACTION D is acceptable because, if not restricted by plant operation, the action of closing the associated MSIVs and MFIVs isolates the affected secondary system line, which accomplishes the safety function of the inoperable MSIS manual and automatic actuation logic channel(s). This action is similar to the ACTIONS provided in ITS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and ITS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)," and may be preferable instead of placing the plant in an unnecessary plant cooldown transient. Alternately, if it is necessary to maintain the MSIVs and MFIVs open to support plant cooldown, the actions continue to allow the plant to be placed in MODE 3 within 6 hours and in MODE 4 within 12 hours consistent with the current time to reach these MODES in CTS 3.0.3 (ITS LCO 3.0.3) and the times used throughout the CTS and ITS to reach MODES 3 and 4. The Completion Time to close the associated MSIVs and MFIVs corresponds to the time to be in MODE 3. These Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. This change is designated as less restrictive because the ITS LCO requirements are applicable in fewer operating conditions than in the CTS.

L02 **Unit 1 only:** *(Category 4 – Relaxation of Required Action)* CTS Table 3.3-3 Functional Unit 1 (SIAS), Functional Unit 2 (CSAS), Functional Unit 3 (CIS), Functional Unit 4 (MSIS) and Functional Unit 5 (RAS) do not specifically include requirements for the automatic actuation logic channels, including actions when one or more automatic actuation logic channels are inoperable. However, CTS Table 4.3-2 list Surveillance Requirements for the ESFAS automatic actuation logic associated with these Functions. Therefore, per the requirements of CTS 4.0.1 (ITS SR 3.0.1), CTS 3.3.2.1 encompasses the ESFAS automatic actuation logic. ITS LCO 3.3.4 requires the two automatic actuation logic channels to be OPERABLE in the MODES specified for each Function. Additionally, ITS 3.3.4 provides ACTIONS for one actuation logic channel inoperable and for two actuation logic channels inoperable. For CTS, if an automatic actuation channel is inoperable, CTS 3.0.3 is entered since there is no Condition and Action specified in the CTS. This changes the CTS by explicitly requiring the two ESFAS actuation logic channels be OPERABLE for each ESAFS Function and providing time to restore an automatic actuation channel to OPERABLE status when one channel (per Function) is inoperable. ITS ACTION C and ACTION E provide a Completion Time of 48 hours or in accordance with the Risk Informed Completion Time Program to restore an inoperable actuation logic channel to OPERABLE status before entry into ACTION D or F, respectively, which provide shutdown requirements. ACTION D requires the MSIVs and MFIVs be closed within 6 hours or be in MODE 3 within 6 hours and MODE 4 within 12 hours. ACTION F requires the unit be in MODE 3 within 6 hours and MODE 5 within 36 hours. This changes the CTS by providing specific ACTIONS to take when an actuation logic channel is inoperable before a shutdown ACTION is entered. CTS 3.0.3 requires the unit to be in MODE 5 within 37 hours from the time that one ESFAS actuation logic channel is discovered inoperable, whereas the ITS

ACTIONS allow a Completion Time of 48 hours or in accordance with the Risk Informed Completion Time Program to restore the inoperable actuation logic channel to OPERABLE status before a shutdown ACTION is required.

The purpose of ITS 3.3.4 ACTIONS C and E is to provide a period of time to restore an inoperable actuation logic channel to OPERABLE status and the purpose of ITS 3.3.4 ACTIONS D and F is to place the unit outside the Applicability of the ESFAS Function with the inoperable actuation logic channel. Currently, an inoperable automatic actuation logic channel requires entry into CTS 3.0.3 since no Action is provided for this case. CTS 3.0.3 allows 1 hour to initiate action, 7 hours for the unit to be placed in MODE 3, and 13 hours for the unit to be placed in MODE 4, and 37 hours for the unit to be in MODE 5.

ITS ACTION C and ACTION E provide a Completion Time of 48 hours or in accordance with the Risk Informed Completion Time Program to restore an inoperable actuation logic channel to OPERABLE status before entry into ACTION D or F, respectively. ACTION D requires the MSIVs and MFIVs closed within 6 hours or be in MODE 3 within 6 hours and MODE 4 within 12 hours. ACTION F requires the unit be in MODE 3 within 6 hours and MODE 5 within 36 hours. The proposed ACTIONS result in extending the time an ESFAS channel may be inoperable, which is less restrictive than the CTS 3.0.3 times of 13 hours for the unit to be placed in MODE 4, and 37 hours for the unit to be in MODE 5.

This change is considered acceptable because the SIAS, CSAS, CIS, MSIS and RAS Functions require the automatic actuation logic channels to operate properly in order to actuate the associated Function response. Two channels are required to be OPERABLE to ensure a single failure of a logic channel does not prevent the actuation of the associated SIAS, CSAS, CIS, MSIS or RAS Function response. The proposed Applicability is consistent with the Applicability of the Functions listed under CTS Table 3.3-3 Functional Units 1.a, 2.a, 3.a, 5.a and 4.a as modified by DOC L01. This change is designated as less restrictive because the ITS Actions are less restrictive than the CTS Actions

L03 (*Category 7 - Relaxation of Surveillance Frequency*) CTS Table 4.3-2 requires CHANNEL FUNCTIONAL TEST requirements for each ESFAS Functional Unit, including ESFAS automatic actuation logic channels. ITS SR 3.3.4.1 also requires a CHANNEL FUNCTIONAL TEST on each ESFAS logic channel and includes a Note (Note 2) that states, “Relays associated with plant equipment that cannot be operated during plant operation are only required to be tested during each MODE 5 entry exceeding 24 hours unless tested during the previous 6 months.” This changes the CTS by providing allowance to test components required to satisfy the CHANNEL FUNCTIONAL TEST at a frequency other than the frequency specified in accordance with the Surveillance Frequency Control Program.

The purpose of the additional Note is to allow certain components required to satisfy the CHANNEL FUNCTIONAL TEST to be tested during plant shutdown conditions. The allowance is necessary to allow for the fact that operating certain relays during power operation could cause plant transients or equipment damage. This change is acceptable because during plant operation actuation of

certain systems and components could result in a plant transient. For example, isolation of the Component Cooling Water System valves to and from the reactor coolant pumps or closure of the MSIVs and feedwater isolation valves. Appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function and assure that the necessary quality of systems and components is maintained, and that facility operation will be within safety limits. Additionally, the relaxation of the testing frequency is offset by reducing the likelihood of a plant transient as a result of testing certain actuating relays during at-power conditions. This change is also consistent with Note 2 in ISTS SR 3.3.5.1. This change is designated as less restrictive because portions of the Surveillance will be performed less frequently under the ITS than under the CTS.

L04 **Unit 2 only**: (*Category 4 – Relaxation of Required Action*) When one MSIS manual trip button channel is inoperable, CTS Table 3.3-3 Action 16 requires the channel to be restored to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5 (i.e., Main Steam Line Isolation Valves Specification). ITS 3.3.4 ACTION C requires restoring a channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program. This changes the CTS by optionally allowing the manual channel to be restored in accordance with the Risk Informed Completion Time (RICT) Program and eliminating the requirement to declare the associated valve inoperable and take ACTION required by the supported system Specification. This change aligns the action for one inoperable Unit 2 MSIS manual channel with the action for one inoperable Unit 1 MSIS manual channel and an inoperable channel for the other ESFAS manual actuation channels.

The purpose of the actions associated with the MSIS manual actuation instrumentation in CTS Table 3.3-3 is to ensure sufficient remedial measures are provided to allow safe operation pursuant the requirements of 10 CFR 50.36(c)(2). Automatic steam line and feedwater line isolation is assumed in the mitigation of a major secondary system pipe rupture accident (e.g., main steam line break (MSLB) or feedwater line break (FWLB)). The consequences of a MSLB inside containment bounds the consequences of a FWLB accident. This change is acceptable because, in part, the MSIS manual actuation instrumentation is not credited in the safety analysis and a safety analysis limit is not specified for this actuation function. This change is also acceptable considering redundant manual actuation logic channels are available to ensure the required equipment supported by the MSIS (i.e., MSIVs and MFIVs) will be capable of actuating, diverse automatic actuation channels are available, and a Completion Time beyond 48 hours will be evaluated, and the risk managed in accordance with the RICT Program. Extending the time to restore a single ESFAS manual actuation channel to OPERABLE status has been previously evaluated and approved by the NRC for the other ESFAS manual channels and the Unit 1 ESFAS manual channels, including the Unit 1 MSIS manual channel. In the NRC safety evaluation (SE) accompanying PSL Unit 2 License Amendment 199 (RICT Amendment), dated July 2, 2019, (NRC ADAMS Accession No. ML19113A099) the NRC concluded that the proposed RICT for the manual ESFAS channels does not impede accomplishing their safety

function and does not alter the existing diversity of the affected functions. The NRC staff stated that the change is consistent with the defense-in-depth philosophy, and therefore acceptable. The acceptability was based, in part, on confirmation that more than one diverse means existed to accomplish the safety function for each identified accident condition, as indicated in the Tables provided in FPL to NRC supplement letter, "Fourth Response to Request for Additional Information Regarding License Amendment Request to Adopt Risk Informed Completion Times TSTF-505, Revision 1, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,'" dated March 15, 2018 (NRC ADAMS Accession No. ML18074A116). Tables 2 and 4 of the supplement letter provided the Instrumentation Diversity for each Unit 1 and Unit 2 ESFAS Instrument Function, respectively. As indicated in Table 4 of the RICT supplement letter, ESFAS diversity for the MSIS manual actuation instrumentation is provided by the Steam Generator Pressure – Low Function for a MSLB outside containment. ESFAS diversity for the MSIS manual actuation instrumentation for a MSLB inside containment is provided by the Steam Generator Pressure – Low Function and the Containment Pressure – High Function. This diversity is equivalent or better than the diversity for the Unit 1 MSIS manual actuation instrumentation, which has been approved to restore one inoperable channel to OPERABLE status in accordance with the RICT Program.

The NRC-approved RICT Program required by CTS 6.8.4.s (ITS 5.5.17) provides controls to calculate a RICT as implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." In the NRC SE accompanying the RICT Amendment, the NRC found that the PSL PRA maintenance and change process ensures that the configuration risk management program models used in the RICT calculations will continue to use PRA methods acceptable to the NRC and that the PRA model will be updated as necessary to reflect the as-built and as-operated plant. The NRC also found that appropriate programmatic and procedural controls for the RICT Program are consistent with the guidance of NEI 06-09, Revision 0-A. Therefore, use of the RICT Program specified in the Technical Specifications to determine a RICT to restore an inoperable MSIS manual actuation channel is acceptable.

Based on the considerations described herein, the change to the actions associated with one inoperable MSIS manual actuation channel continues to provide sufficient remedial measures to allow safe operation pursuant the requirements of 10 CFR 50.36(c)(2). This change is designated as less restrictive because the ITS Actions are less restrictive than the CTS Actions.

L05 **Unit 1 only**: (*Category 4 – Relaxation of Required Action*) CTS Table 3.3-3 Functional Unit 1 (SIAS), Functional Unit 2 (CSAS), Functional Unit 3 (CIS), Functional Unit 4 (MSIS) and Functional Unit 5 (RAS) do not specifically include requirements for the automatic actuation logic channels, including actions when one or more automatic actuation logic channels are inoperable. However, CTS Table 4.3-2 list Surveillance Requirements for the ESFAS automatic actuation logic associated with these Functions. Therefore, per the requirements of CTS 4.0.1 (ITS SR 3.0.1), CTS 3.3.2.1 encompasses the ESFAS automatic actuation logic. ITS LCO 3.3.4 requires the two automatic actuation logic channels to be OPERABLE in the MODES specified for each Function. Additionally, ITS 3.3.4

provides ACTIONS for inoperable actuation logic channels. For CTS, if an automatic actuation channel is inoperable, CTS 3.0.3 is entered since there is no Condition and Action specified in the CTS. When one MSIS automatic logic channel is inoperable, ITS 3.3.4 ACTION C requires restoring a channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time (RICT) Program. When one automatic logic channel associated with one or more ESFAS Functions (except MSIS and AFAS) is inoperable, ITS 3.3.4 ACTION E also requires restoring a channel to OPERABLE status within 48 hours or in accordance with the RICT Program. This changes the CTS by providing time to restore one automatic logic channel of each ESFAS actuation Function within 48 hours or in accordance with the RICT Program. This change aligns the action for one inoperable SIAS, CSAS, CIS, MSIS, and RAS automatic actuation logic channel with the action for one inoperable AFAS automatic actuation logic channel, one inoperable ESFAS manual channel, and an inoperable channel for the Unit 2 ESFAS automatic logic actuation channels.

The purpose of the actions associated with the SIAS, CSAS, CIS, MSIS, and RAS automatic actuation instrumentation in CTS Table 3.3-3 is to ensure sufficient remedial measures are provided to allow safe operation pursuant the requirements of 10 CFR 50.36(c)(2). The SIAS, CSAS, CIS, MSIS, and RAS automatic actuation logic channels are implicitly assumed to be functional to support automatic actuation of the supported ESF equipment during design basis accidents and transients. This change is acceptable considering redundant automatic actuation logic channels are available to ensure the required equipment supported by the ESFAS (e.g., ECCS and Containment Spray System) will be capable of automatically actuating, diverse manual actuation channels are available, and a Completion Time beyond 48 hours will be evaluated, and the risk managed in accordance with the RICT Program. Extending the time to restore a single ESFAS automatic actuation logic channel to OPERABLE status has been previously evaluated and approved by the NRC for one inoperable AFAS automatic actuation logic channel, the ESFAS manual channels, and the Unit 2 ESFAS automatic actuation logic channels. In the NRC safety evaluation (SE) accompanying PSL Unit 1 License Amendment 247 (RICT Amendment), dated July 2, 2019, (NRC ADAMS Accession No. ML19113A099) the NRC concluded that the proposed RICT for the manual ESFAS channels and the Unit 2 automatic actuation logic channels does not impede accomplishing their safety function and does not alter the existing diversity of the affected functions. The NRC staff stated that the change is consistent with the defense-in-depth philosophy, and therefore acceptable. The acceptability was based, in part, on confirmation that more than one diverse means existed to accomplish the safety function for each identified accident condition, as indicated in the tables provided in FPL to NRC supplement letter, “Fourth Response to Request for Additional Information Regarding License Amendment Request to Adopt Risk Informed Completion Times TSTF-505, Revision 1, ‘Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,’” dated March 15, 2018 (NRC ADAMS Accession No. ML18074A116). Tables 2 and 4 of the supplement letter provided the Instrumentation Diversity for each Unit 1 and Unit 2 ESFAS Instrument Function, respectively. As indicated in Table 2 of the RICT supplement letter, ESFAS diversity for the SIAS, CSAS, CIS, MSIS, and RAS automatic actuation instrumentation is provided by the respective manual

actuation channels for each specified accident condition. Equivalent actions in Unit 2 CTS (Table 3.3-3, Action 12) allow restoration of one inoperable automatic actuation channel to OPERABLE status in 48 hours or in accordance with the RICT Program.

The NRC-approved RICT Program required by CTS 6.8.4.r (ITS 5.5.17) provides controls to calculate a RICT as implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." In the NRC SE accompanying the RICT Amendment, the NRC found that the PSL PRA maintenance and change process ensures that the configuration risk management program models used in the RICT calculations will continue to use PRA methods acceptable to the NRC and that the PRA model will be updated as necessary to reflect the as-built and as-operated plant. The NRC also found that appropriate programmatic and procedural controls for the RICT Program are consistent with the guidance of NEI 06-09, Revision 0-A. Therefore, use of the RICT Program specified in the Technical Specifications to determine a RICT to restore an inoperable SIAS, CSAS, CIS, MSIS, or RAS automatic actuation logic channel is acceptable.

Based on the considerations described herein, the change to the Unit 1 CTS actions associated with one inoperable SIAS, CSAS, CIS, MSIS, or RAS automatic actuation logic channel continues to provide sufficient remedial measures to allow safe operation pursuant the requirements of 10 CFR 50.36(c)(2). This change is designated as less restrictive because the ITS Actions are less restrictive than the CTS Actions.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.3 INSTRUMENTATION (Analog)

Actuation

3.3.2.1 3.3.~~5~~ Engineered Safety Features Actuation System (ESFAS) Logic and Manual ~~Trip~~ ~~(Analog)~~

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Actuation

3.3.2.1 LCO 3.3.~~5~~ Two ESFAS Manual ~~Trip~~ and two ESFAS Actuation Logic channels shall be OPERABLE for each ESFAS Function specified in Table 3.3.~~5~~-1.

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Table 3.3-3 Functional Unit 1.a, 2.a, 3.a, 4.a, 5.a., 7.a, 7.b APPLICABILITY: According to Table 3.3.~~5~~-1.

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ACTIONS

----------NOTE----------

DOC A06 Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-9 Action 11 A. One or more Functions with one Auxiliary Feedwater Actuation Signal (AFAS) Manual ~~Trip~~ or Actuation Logic channel inoperable. Actuation	A.1 Restore channel to OPERABLE status.	48 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
Table 3.3-9 Action 11 DOC A04 B. Two AFAS Manual ~~Trip~~ or Actuation Logic channels inoperable. Actuation OR Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours

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2

INSERT 1

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1

INSERT 1

	CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-9 Action 8 DOC L01 DOC L02 DOC L05	C. One Main Steam Isolation System (MSIS) Manual Actuation or Actuation Logic channel inoperable.	C.1 Restore channel to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program
Table 3.3-9 Action 8 DOC A03 DOC L01 DOC L02	D. Two MSIS Manual Actuation or Actuation Logic channels inoperable. OR Required Action and associated Completion Time of Condition C not met.	D.1.1 Close main steam isolation valves. AND D.1.2 Close main feedwater isolation valves. OR D.2.1 Be in MODE 3. AND D.2.2 Be in MODE 4.	6 hours 6 hours 6 hours 12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-9 Action 8 DOC L02 ~~C~~. One or more Functions with one Manual ~~Trip~~ or Actuation Logic channel inoperable except AFAS. and MSIS (E)	~~C~~.1 Restore channel to OPERABLE status. (E)	48 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
~~D. Required Action and associated Completion Time of Condition C not met.~~	~~D.1 Be in MODE 3.~~ ~~AND~~ ~~D.2 NOTE LCO 3.0.4.a is not applicable when entering MODE 4.~~ ~~Be in MODE 4.~~	~~6 hours~~ ~~12 hours~~
DOC A04 DOC L02 ~~E~~. One or more Functions with two Manual ~~Trip~~ or Actuation Logic channel inoperable except AFAS. and MSIS (F) OR Required Action and associated Completion Time of Condition E not met.	~~E~~.1 Be in MODE 3. (F) AND ~~E~~.2 Be in MODE 5. (F)	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
Table 4.3-2 Functional Unit 1.d, 2.c, 3.d, 4.c, 5.c, 7.c DOC L03	SR 3.3.~~5~~.1 (4)	----------NOTES---------- 1. Testing of Actuation Logic shall include verification of the proper operation of each initiation relay. 2. Relays associated with plant equipment that cannot be operated during plant operation are only required to be tested during each MODE 5 entry exceeding 24 hours unless tested during the previous 6 months. --- Perform a CHANNEL FUNCTIONAL TEST on each ESFAS logic channel.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
Table 4.3-2 Functional Unit 1.a, 2.a, 3.a, 4.a, 5.a, 7.a	SR 3.3.~~5~~.2 (4)	Perform a CHANNEL FUNCTIONAL TEST on each ESFAS Manual ~~Trip~~ (Actuation) channel.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

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Table 3.3.~~5~~-1 (page 1 of 1)
Engineered Safety Features Actuation System Actuation Logic and
Manual Channel Applicability

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FUNCTION	APPLICABLE MODES
1. Safety Injection Actuation Signal	1,2,3,~~[4]~~
2. Containment Spray Actuation Signal	1,2,3,~~[4]~~
3. Containment Isolation Actuation Signal	1,2,3,4
4. Main Steam Isolation Signal	1,2,3,~~4~~ (a)
5. Recirculation Actuation Signal	1,2,3,4
6. Auxiliary Feedwater Actuation Signal	1,2,3

DOC L01

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DOC L01

(a) except when valves isolated by the MSIS Function are closed

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3.3 INSTRUCTION (Analog)

3.3.2 3.3.~~5~~ Engineered Safety Features Actuation System (ESFAS) Logic and Manual ~~Trip~~ ~~(Analog)~~

3.3.2 LCO 3.3.~~5~~ Two ESFAS Manual ~~Trip~~ and two ESFAS Actuation Logic channels shall be OPERABLE for each ESFAS Function specified in Table 3.3.~~5~~-1.

Table 3.3-3 Functional Unit 1.a, 1.d, 2.a, 2.c, 3.a, 3.e, 4.a, 4.d, 5.a., 5.c, 7.a, 7.b APPLICABILITY: According to Table 3.3.~~5~~-1.

ACTIONS

----------NOTE----------

DOC A06 Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-9 Action 15 A. One or more Functions with one Auxiliary Feedwater Actuation Signal (AFAS) Manual ~~Trip~~ or Actuation Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
Table 3.3-9 Action 15 DOC A04 B. Two AFAS Manual ~~Trip~~ or Actuation Logic channels inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours

INSERT 1

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INSERT 1

	CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-9 Action 16 DOC L01 DOC L04	C. One Main Steam Isolation System (MSIS) Manual Actuation or Actuation Logic channel inoperable.	C.1 Restore channel to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program
Table 3.3-9 Action 12 DOC A03 DOC L01	D. Two MSIS Manual Actuation or Actuation Logic channels inoperable. OR Required Action and associated Completion Time of Condition C not met.	D.1.1 Close main steam isolation valves. AND D.1.2 Close main feedwater isolation valves. OR D.2.1 Be in MODE 3. AND D.2.2 Be in MODE 4.	6 hours 6 hours 6 hours 12 hours

CTS

Actuation

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-9 Action 12 ~~C~~. One or more Functions with one Manual ~~Trip~~ or Actuation Logic channel inoperable except AFAS. and MSIS E	~~C~~.1 Restore channel to OPERABLE status. E	48 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
~~D. Required Action and associated Completion Time of Condition C not met.~~	~~D.1~~ ~~Be in MODE 3.~~ ~~AND~~ ~~D.2~~ ~~NOTE LCO 3.0.4.a is not applicable when entering MODE 4.~~ ~~Be in MODE 4.~~	~~6 hours~~ ~~12 hours~~
DOC A04 ~~E~~. One or more Functions with two Manual ~~Trip~~ or Actuation Logic channel inoperable except AFAS. and MSIS F OR Required Action and associated Completion Time of Condition E not met.	~~E~~.1 Be in MODE 3. F AND ~~E~~.2 Be in MODE 5. F	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
Table 4.3-2 Functional Unit 1.d, 2.c, 3.e, 4.d, 5.c, 7.c DOC L03 SR 3.3.~~5~~.1 4	----------NOTES---------- 1. Testing of Actuation Logic shall include verification of the proper operation of each initiation relay. 2. Relays associated with plant equipment that cannot be operated during plant operation are only required to be tested during each MODE 5 entry exceeding 24 hours unless tested during the previous 6 months. ---------- Perform a CHANNEL FUNCTIONAL TEST on each ESFAS logic channel.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2
Table 4.3-2 Functional Unit 1.a, 2.a, 3.a, 4.a, 5.a, 7.a SR 3.3.~~5~~.2 4	Perform a CHANNEL FUNCTIONAL TEST on each ESFAS Manual ~~Trip~~ Actuation channel.	~~[[18] months~~ 3 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

CTS

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3

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Table 3.3.~~5~~-1 (page 1 of 1)
Engineered Safety Features Actuation System Actuation Logic and
Manual Channel Applicability

FUNCTION	APPLICABLE MODES
1. Safety Injection Actuation Signal	1,2,3,~~[4]~~
2. Containment Spray Actuation Signal	1,2,3,~~[4]~~
3. Containment Isolation Actuation Signal	1,2,3,4
4. Main Steam Isolation Signal	1,2,3,~~4~~ (a)
5. Recirculation Actuation Signal	1,2,3,4
6. Auxiliary Feedwater Actuation Signal	1,2,3

DOC L01

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DOC L01

(a) except when valves isolated by the MSIS Function are closed

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JUSTIFICATION FOR DEVIATIONS
ITS 3.3.4, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) LOGIC AND MANUAL ACTUATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The heading for ISTS 3.3.5 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2). Therefore, ISTS 3.3.2 is not included in the PSL ITS. The successive Specifications are renumbered as necessary.

4. ISTS Table 3.3.5-1, Function 4, Main Steam Isolation Signal, specifies an Applicability of MODES 1, 2, 3, and 4. ITS 3.3.4-1 (ISTS 3.3.5-1) requires the same function in MODES 1, 2, and 3 consistent with Unit 2 CTS for the same function. MODES 2 and 3 are also modified by Footnote (a) that states, “except when valves isolated by the MSIS Function are closed.” As a result of the change to the Applicability, ITS 3.3.4 includes proposed ACTIONS C and D. When one MSIS Manual Actuation or Actuation Logic channels is inoperable, proposed ACTION C requires restoring the channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program consistent with the Unit 1 CTS Actions. Discussion of Change L04 provides justification for the change to the Completion Time to restore the manual channel to OPERABLE status in the Unit 2 CTS Actions. When two MSIS Manual Actuation or Actuation Logic channels are inoperable or when the action to restore one MSIS Manual Actuation or Actuation Logic channel cannot be completed within the required Completion Time, proposed ACTION D requires closing the main steam isolation valves and the main feedwater isolation valves in 6 hours, or be in MODE 3 in 6 hours and MODE 4 in 12 hours.

   This ISTS deviation aligns the Applicability of the MSIS actuation instrumentation with the applicability requirements of the associated MSIS instrument channels (i.e., Footnote (f) to ISTS 3.3.4-1 (ITS 3.3.3-1), Function 4.a) and similar to the applicability of the supported equipment Specifications ITS 3.7.2, “Main Steam Isolation Valves (MSIVs)” and ITS 3.7.4, “Main Feedwater Isolation Valves (MFIVs).” Proposed ITS 3.3.4 ACTION C is added to differentiate the ACTIONS for the MSIS from the ACTIONS of the other ESFAS Logic and Manual Actuation signals. ITS 3.3.4 ACTION D is added to provide appropriate actions to reflect the change to the Applicability, which provides sufficient remedial actions to safely continue operation in lieu of a plant cooldown. Discussion of Change L01 provides justification for the change to the CTS Applicability for the MSIS and proposed actions to close the MSIVs and MFIVs.

   The purpose of the MSIS in ISTS Table 3.3.5-1 (ITS 3.3.4-1) is to ensure the ESFAS MSIS Functions are OPERABLE when the supported equipment is required to be

OPERABLE. The exception to MODE 2 and 3 is added to clarify that the MSIS manual and automatic actuation instrumentation is not required to be OPERABLE when the valves actuated by the MSIS instrumentation are in the closed position. Proposed ACTION D places the unit in a MODE or other specified condition in which the LCO does not apply by either closing the MSIVs and MFIVs within 6 hours (proposed Required Actions D.1.1 and D.1.2) or in MODE 3 within 6 hours and in MODE 4 within 12 hours (proposed Required Actions D.2.1 and D.2.2). The Completion Time of Required Actions D.1.1 and D.1.2 is consistent with the Completion Time to be in MODE 3. The 6-hour and 12-hour Completion Times of Required Actions D.2.1 and D.2.2, respectively, are based on the Completion Times to reach MODES 3 and 4 elsewhere in the ISTS and are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. Subsequent Conditions and Required Actions are renumerated, as applicable.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~5~~ 4 Engineered Safety Features Actuation System (ESFAS) Logic and Manual ~~Trip~~ Actuation ~~(Analog)~~

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary and to mitigate accidents.

The ESFAS contains devices and circuitry that generate the following signals when the monitored variables reach levels that are indicative of conditions requiring protective action:

1. Safety Injection Actuation Signal (SIAS),
2. Containment Spray Actuation Signal (CSAS),
3. Containment Isolation ~~Actuation~~ Signal (CI~~A~~S),
4. Main Steam Isolation Signal (MSIS),
5. Recirculation Actuation Signal (RAS), and
6. Auxiliary Feedwater Actuation Signal (AFAS).

Equipment actuated by each of the above signals is identified in the FSAR (Ref. 1).

Each of the above ESFAS actuation systems is segmented into four sensor subsystems addressed by LCO 3.3.~~4~~ 5, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," and two actuation subsystems addressed by this LCO. Each sensor subsystem includes measurement channels and bistables. The SIAS actuation subsystems include two logic subsystems for sequentially loading the diesel generators.

Each of the four sensor subsystem channels monitors redundant and independent process measurement channels. Each sensor is monitored by at least one bistable. The bistable associated with each ESFAS Function will trip when the monitored variable exceeds the trip setpoint. When tripped, the sensor subsystems provide outputs to the two actuation subsystems.

BASES

BACKGROUND (continued)

The two independent actuation subsystems each compare the four associated sensor subsystem outputs. If a trip occurs in two or more sensor subsystem channels, the two-out-of-four logic in each actuation subsystem will initiate one train of ESFAS. Each has sufficient equipment to provide protection to the public in the case of a Design Basis Event. The sensor subsystem is addressed in LCO 3.3.~~4~~ 3. This LCO addresses the actuation subsystem. 3

Each of the four sensor subsystems is mounted in a separate cabinet, excluding the sensors and field wiring.

The role of the sensor subsystem (measurement channels and bistables) is discussed in LCO 3.3.~~4~~ 3. That of the actuation subsystem is discussed below. 3

ESFAS Logic

The two independent actuation subsystems compare the four sensor subsystem outputs. If a trip occurs in the same parameter in two or more sensor subsystem channels, the two-out-of-four logic in each actuation subsystem initiates one train of ESFAS. Either train controls sufficient redundant and independent equipment.

INSERT 1

Each actuation subsystem channel is housed in two cabinets. One cabinet contains the logic circuitry for the actuation channel, while the other cabinet contains the power relay equipment. This power relay equipment includes the power relays (initiation relays) that actuate the ESFAS equipment in response to a signal from the Actuation Logic.

It is possible to change the two-out-of-four ESFAS Logic to a two-out-of-three logic for a given input parameter in one channel at a time by disabling one channel input to the logic. Thus, the bistables will function normally, producing normal trip indication and annunciation, but ESFAS actuation will not occur since the bypassed channel is effectively removed from the coincidence logic. Maintenance bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. ~~At some plants an interlock prevents simultaneous maintenance bypassing of the same parameter in more than one channel.~~ Maintenance bypassing is normally employed during maintenance or testing. 1

1

INSERT 1

Each ESFAS consists of four measurement channels (designated MA, MB, MC, and MD) for each input parameter, two logic matrix systems (SA and SB) and two actuation channels (A and B).

Each measurement channel consists of a sensor, power supply and bistable unit arranged in a current flow loop circuit. The bistable unit provides a digital signal to logic matrices where signals from all four measurement channels for that parameter are combined in a two-out-of-four logic network. Isolation devices are provided to maintain separation between the measurement channels and the logic matrices. Logic matrices (SA or SB) provide initiation signals to their associated actuation channels (A or B) when the logic for the particular signal is satisfied.

BASES

BACKGROUND (continued)

~~For plants that have demonstrated sufficient channel to channel independence,~~ two-out-of-three logic is the minimum that is required to provide adequate plant protection, since a failure of one channel still ensures that ESFAS actuation would be generated by the two remaining OPERABLE channels. Two-out-of-three logic also prevents inadvertent actuation caused by any single channel failure in a trip condition. 1

In addition to the maintenance bypasses, there are operating bypasses (blocks) on the Pressurizer Pressure - Low input to the SIAS and on the Steam Generator Pressure - Low input to the MSIS when these inputs are no longer required for protection. These bypasses are enabled manually when the enabling conditions are satisfied in three of the four sensor subsystem channels. The operating bypass circuitry employs four bistable channels in the sensor subsystems, sensing pressurizer pressure (for the SIAS) and steam generator pressure (for the MSIS). These bistables provide contact output to the three-out-of-four logic in the two actuation subsystem channels. When the logic is satisfied, manual bypassing is permitted. There are two manual bypass actuation controls for each Function, one per train.

All operating bypasses are automatically removed when enabling bypass conditions are no longer satisfied.

Manual ESFAS initiation capability is provided to permit the operator to manually actuate an Engineered Safety Features (ESF) System when necessary. Two push buttons are provided in the control room for each ESFAS Function. Each push button actuates one train via the ESFAS Logic.

The Actuation Logic is tested by inserting a local test signal. A coincidence logic trip will occur if there is the simultaneous presence of a sensor channel trip, either legitimate or due to testing. Most ESFAS Functions employ several separate parallel two-out-of-four Actuation Logic modules, with each module actuating a subset of the ESFAS equipment associated with that Function. Each of these subchannels can be tested individually so that simultaneous actuation of an entire train can be avoided during testing.

INSERT 2 → ~~Except in the case of actuation subchannels SIAS Nos. 5 and 10, CIAS No. 5, and MSIS No. 1, all Actuation Logic channels can be tested at power. The above designated subchannels must be tested when shut down because they actuate the following equipment, which cannot be actuated at power:~~ 1

1

INSERT 2

Some logic channels are testable during reactor operation without affecting the operability or safety of the plant. These devices are tested by imposing an ESFAS during reactor operation. Other logic channels are not testable during reactor operation and must be tested when shutdown to prevent affecting the operability or safety of the plant. Testing Criteria are identified in the UFSAR (Ref. 1).

BASES

BACKGROUND (continued)

- ~~Reactor coolant pump (RCP) seal bleedoff isolation valves,~~
- ~~Service water isolation valves,~~
- ~~Volume control tank (VCT) discharge valves,~~
- ~~Letdown stop valves,~~
- ~~Component cooling water (CCW) to RCPs,~~
- ~~CCW from RCPs,~~
- ~~Main steam isolation valves (MSIVs),~~
- ~~Feedwater isolation valves, and~~
- ~~Instrument air containment isolation valves.~~

1

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. Functions such as Manual Initiation, not specifically credited in the accident analysis, serve as backups to Functions and are part of the NRC staff approved licensing basis for the plant.

ESFAS protective Functions are as follows:

1. Safety Injection Actuation Signal

   The SIAS ensures acceptable consequences during loss of coolant accident (LOCA) events, including steam generator tube rupture, and main steam line breaks (MSLBs) or feedwater line breaks (FWLBs) (inside containment). To provide the required protection, either a high containment pressure or a low pressurizer pressure signal will initiate SIAS. SIAS initiates the Emergency Core Cooling Systems (ECCS) and performs several other Functions, such as initiating control room isolation and starting the diesel generators.

BASES

APPLICABLE SAFETY ANALYSES (continued)

2. Containment Spray Actuation Signal

The CSAS initiates containment spray, preventing containment overpressurization during a LOCA or MSLB. ~~At some plants,~~ both a high containment pressure signal and an SIAS have to actuate to provide the required protection. This configuration reduces the likelihood of inadvertent containment spray. 1

3. Containment Isolation ~~Actuation~~ Signal 1

The CI~~A~~S actuates the Containment Isolation System, ensuring acceptable consequences during LOCAs and MSLBs or FWLBs (inside containment). To provide protection, a high containment pressure signal will initiate CI~~A~~S at the same setpoint at which an SIAS is initiated. 1 1

4. Main Steam Isolation Signal

The MSIS ensures acceptable consequences during an MSLB or FWLB by isolating both steam generators if either generator indicates a low steam generator pressure. The MSIS, concurrent with or following a reactor trip, minimizes the rate of heat extraction and subsequent cooldown of the RCS during these events.

5. Recirculation Actuation Signal

At the end of the injection phase of a LOCA, the refueling water tank (RWT) will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from RWT to containment sump must occur before the RWT empties to prevent damage to the ECCS pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support pump suction. Furthermore, early switchover must not occur to ensure sufficient borated water is injected from the RWT to ensure the reactor remains shut down in the recirculation mode. An RWT Level - Low signal initiates the RAS.

BASES

APPLICABLE SAFETY ANALYSES (continued)

6. Auxiliary Feedwater Actuation Signal

the affected

An AFAS initiates feedwater flow to ~~both~~ steam generator~~s~~ if a low level is indicated in ~~either~~ steam generator, unless the generator is ~~ruptured~~.

the affected

faulted

The AFAS maintains a steam generator heat sink during the following events:

- MSLB,
- FWLB,
- Inadvertent opening of a steam generator atmospheric dump valve, and
- Loss of feedwater.

A low steam generator water level signal will initiate auxiliary feed to the affected steam generator.

INSERT 3

1

Secondary steam generator (SG) differential pressure (SG-A > SG-B) or (SG-B > SG-A) inhibits auxiliary feed to a generator identified as being ~~ruptured~~. This input to the AFAS logic prevents loss of the intact generator while preventing feeding a ~~ruptured~~ generator during MSLBs and FWLBs. This prevents containment overpressurization during these events.

faulted

faulted

INSERT 4

1

The ESFAS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that all components necessary to provide an ESFAS actuation be OPERABLE.

Actions allow maintenance bypass of individual channels. Plants are restricted to 48 hours in a maintenance bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic).

The Bases for the LCO on ESFAS automatic actuation Functions are addressed in the Bases for LCO 3.3.~~4~~. Those associated with the Manual ~~Trip~~ or Actuation Logic are addressed below.

3

Actuation

3

1

INSERT 3

A separate auxiliary feedwater actuation signal is generated for each steam generator (AFAS-1, AFAS-2). For each AFAS-1 and AFAS-2 there are four independent level transmitters for steam generator level, four independent steam generator pressure transmitters, and four independent feedwater header pressure transmitters. The AFAS actuation logic actuates auxiliary feedwater to a steam generator on low level after a time delay period unless that steam generator or its associated auxiliary feedwater supply header have been identified as being faulted. R1

1

INSERT 4

The AFAS actuation logic isolates auxiliary feedwater flow to a steam generator upon recovery of steam generator level.

BASES

LCO (continued)

1. Safety Injection Actuation Signal

a. Manual ~~Trip~~ Actuation

1

This LCO requires two channels of SIAS Manual Trip Actuation to be OPERABLE in MODES 1, 2, 3, and 4.

1

b. Actuation Logic

This LCO requires two channels of SIAS Actuation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

~~Failures in the actuation subsystems, including the manual bypass key switches, are Actuation Logic failures and are addressed in this LCO.~~

1

Actuation Logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

2. Containment Spray Actuation Signal

CSAS is initiated either manually or automatically. ~~At many plants~~ it is also necessary to have an automatic or manual SIAS for a complete actuation. ~~The SIAS opens the containment spray valves, whereas the CSAS actuates other required components.~~ The SIAS requirement should always be satisfied on a legitimate CSAS, since the Containment Pressure - High signal used in the SIAS ~~is the same~~ setpoint used in the CSAS. ~~The transmitters used to initiate CSAS are independent of those used in the SIAS to prevent inadvertent containment spray due to failures in two sensor channels.~~

1

1

1

setpoint is less than the Containment Pressure – High High

a. Manual ~~Trip~~ Actuation

1

Actuation

This LCO requires two channels of CSAS Manual ~~Trip~~ to be OPERABLE in MODES 1, 2, 3, and 4.

b. Actuation Logic

This LCO requires two channels of CSAS Actuation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

Actuation Logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

BASES

LCO (continued)

3. Containment Isolation ~~Actuation~~ Signal

a. Manual ~~Trip~~ (Actuation) 1

This LCO requires two channels of CI~~A~~S Manual ~~Trip~~ (Actuation) to be OPERABLE in MODES 1, 2, 3, and 4. 1

b. Actuation Logic

This LCO requires two channels of Actuation Logic for CI~~A~~S to be OPERABLE in MODES 1, 2, 3, and 4. 1

Actuation Logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

4. Main Steam Isolation Signal

a. Manual ~~Trip~~ (Actuation) 1

This LCO requires two channels ~~per steam generator~~ of the MSIS Manual ~~Trip~~ (Actuation) to be OPERABLE in MODES 1, 2, 3~~, and 4~~ (and except when all MSIVs are closed). 1

Each channel actuates its associated MSIS Actuation Logic channel. Actuating either MSIS Manual Actuation will result in a full main steam isolation.

b. Actuation Logic 1

This LCO requires two channels of MSIS Actuation Logic to be OPERABLE in MODE~~S~~ 1, 2~~,~~ 3~~, and 4~~ (and MODES and except when all MSIVs are closed). 1

Each MSIS Actuation Logic channel provides a signal to its respective MSIV. An interposing relay will cause the opposite train MSIV to close.

~~Failures in the actuation subsystems, including the manual bypass key switches, are considered Actuation Logic failures and are addressed in the logic LCO.~~

5. Recirculation Actuation Signal

a. Manual ~~Trip~~ (Actuation) 1

This LCO requires two channels of RAS Manual ~~Trip~~ (Actuation) to be OPERABLE in MODES 1, 2, 3, and 4. 1

b. Actuation Logic

This LCO requires two channels of RAS Actuation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

BASES

LCO (continued)

6. Auxiliary Feedwater Actuation Signal

the affected steam
steam

A low level in ~~either~~ generator, as sensed by a two-out-of-four coincidence of four wide range sensors for each generator, will its generate an auxiliary feedwater actuation signal (AFAS), which starts the affected ~~both~~ train~~s~~ of auxiliary feedwater (AFW) pumps and feeds ~~both~~ steam generator~~s~~. The AFAS also monitors the secondary differential pressure in both steam generators and initiates an AFAS block signal to a ~~ruptured~~ generator if the pressure in that generator is lower than the other generator by the differential pressure setpoint.

faulted

a. Manual ~~Trip~~

Actuation

This LCO requires two channels of AFAS Manual ~~Trip~~ to be OPERABLE in MODES 1, 2, and 3.

Actuation

b. Actuation Logic

This LCO requires two channels of AFAS Actuation Logic to be OPERABLE in MODES 1, 2, and 3.

Actuation Logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

1
1
1
1
1

APPLICABILITY All ESFAS Functions are required to be OPERABLE in MODES 1, 2, and 3. In MODES 1, 2, and 3, there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the MSIVs to preclude a positive reactivity addition,
- Actuate AFW to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available),
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or MSLB, and
- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

BASES

APPLICABILITY (continued)

In MODES 4, 5, and 6, automatic actuation of ESFAS Functions is not required, because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating the ESF components if required. ESFAS Manual ~~Trip~~ capability is required for Functions other than AFAS and MSIS in MODE 4 even though automatic actuation is not required. Because of the large number of components actuated on each ESFAS, actuation is simplified by the use of the Manual ~~Trip~~ push buttons. Manual ~~Trip~~ of AFAS is not required in MODE 4 because AFW or shutdown cooling will already be in operation in this MODE.

The ESFAS Actuation Logic must be OPERABLE in the same MODES as the Automatic and Manual ~~Trips~~. In MODE 4, only the portion of the ESFAS logic responsible for the required Manual ~~Trip~~ must be OPERABLE.

In MODES 5 and 6, ESFAS initiated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components.

ACTIONS

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered, if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.~~5~~-1 in the LCO. Completion Times for the inoperable channel of a Function will be tracked separately.

A.1

Condition A applies to one AFAS Manual ~~Trip~~ or AFAS Actuation Logic channel inoperable. It is ~~identical~~ to Condition ~~C~~ for the other ESFAS Functions, except for the shutdown track imposed by Condition ~~D~~.

The channel must be restored to OPERABLE status to restore redundancy of the AFAS Function. The 48 hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~

BASES

ACTIONS (continued)

B.1 and B.2

Actuation

If two Manual ~~Trip~~ or Actuation Logic channels are inoperable or the Required Action and associated Completion Time of Condition A cannot be met, the reactor should be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within ~~[~~12~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

1
2

INSERT 5

1

~~C~~.1

E
E
F
Actuation
and MSIS

Condition ~~C~~ applies to one Manual ~~Trip~~ or Actuation Logic channel inoperable for those ESFAS Functions that must be OPERABLE in MODES 1, 2, 3, and 4 (all Functions except AFAS). The shutdown track imposed by Condition ~~D~~ requires entry into MODE 5, where the LCO does not apply to the affected Functions. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~

3
1
2

The channel must be restored to OPERABLE status to restore redundancy of the affected Functions. The 48 hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel.

~~D.1 and D.2~~

1

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

2

1

INSERT 5

C.1

Condition C applies when one MSIS Manual Actuation or Actuation Logic channel is inoperable.

The 48 hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the remaining OPERABLE MSIS Manual Actuation or Actuation Logic channel. Alternatively, the Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

D.1.1, D.1.2, D.2.1, and D.2.2

If both MSIS Manual Actuation or Actuation Logic channels are inoperable or if the Required Actions and associated Conditions of Condition C are not met, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by closing the MSIVs and MFIVs within 6 hours (Required Actions D.1.1 and D.1.2). If allowed, operation in MODES 2 or 3 with MSIVs and MFIVs closed may continue because isolating the affected main steam line and main feedwater line accomplishes the safety function of the inoperable channel. Alternately, if the MSIVs and MFIVs cannot be closed, the plant must be placed in MODE 3 within 6 hours and in MODE 4 within 12 hours (Required Actions D.2.1 and D.2.2).

The Completion Times of Required Actions D.1.1 and D.1.2 is consistent with the Completion Time to be in MODE 3 for similar conditions and correspond to the Completion Time of D.2.1. The Completion Times of Required Actions D.2.1 and D.2.2 are based on the Completion Time to reach MODE 3 (i.e., 6 hours) and allow an additional 6 hours to reach MODE 4, respectively, if the MSIVs and MFIVs are not closed within 12 hours. These Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

~~Condition D is entered when the Required Action and associated Completion Time of Condition C is not met. The plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.~~ 1

~~Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

~~Required Action D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~

F

F F

~~E~~.1 and ~~E~~.2

Condition ~~E~~ is entered when one or more Functions have two Manual ~~Trip~~ or Actuation Logic channels inoperable except AFAS. The plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

and MSIS

Actuation 3

1

SURVEILLANCE REQUIREMENTS

SR 3.3.~~5~~.1

4

A CHANNEL FUNCTIONAL TEST is performed to ensure the entire channel will perform its intended function when needed. Sensor subsystem tests are addressed in LCO 3.3.4. This SR addresses Actuation Logic tests. A successful test of the required contact(s) of a

3

4

SURVEILLANCE REQUIREMENTS (continued)

channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Actuation Logic Tests

Actuation subsystem testing includes injecting one trip signal into each two-out-of-four logic subsystem in each ESFAS Function and using a bistable trip input to satisfy the trip logic. Initiation relays associated with the affected channel will then actuate the individual ESFAS components. Since each ESFAS Function employs subchannels of Actuation Logic, it is possible to actuate individual components without actuating an entire ESFAS Function.

Note 1 requires that Actuation Logic tests include operation of initiation relays. Note 2 allows deferred at power testing of certain relays to allow for the fact that operating certain relays during power operation could cause plant transients or equipment damage. ~~Those initiation relays that cannot be tested at power must be tested in accordance with Note 2. These include [SIAS No. 5, SIAS No. 10, CIAS No. 5, and MSIS No. 1.]~~ 1 2

~~These relays actuate the following components, which cannot be tested at power:~~

- ~~RCP seal bleedoff isolation valves,~~
- ~~Service water isolation valves,~~ 1
- ~~VCT discharge valves,~~
- ~~Letdown stop valves,~~
- ~~CCW to and from the RCPs,~~
- ~~MSIVs and feedwater isolation valves, and~~
- ~~Instrument air containment isolation valves.~~

~~The reasons that each of the above cannot be fully tested at power are stated in Reference 1.~~

These tests verify that the ESFAS is capable of performing its intended function, from bistable input through the actuated components.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[The Frequency of [92] days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 3).~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

SR 3.3.~~5~~.2
4

Actuation

A CHANNEL FUNCTIONAL TEST is performed on the manual ESFAS actuation circuitry, de-energizing relays and providing Manual ~~Trip~~ of the Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Actuation

This Surveillance verifies that the trip push buttons are capable of opening contacts in the Actuation Logic as designed, de-energizing the initiation relays and providing Manual ~~Trip~~ of the Function. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at a Frequency of once every [18] months.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

REFERENCES

1. FSAR, Section ~~[~~7.3~~]~~. U 1 2

~~2. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.~~ 1

~~3. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989.~~ 1

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~5~~ Engineered Safety Features Actuation System (ESFAS) Logic and Manual ~~Trip~~ ~~(Analog)~~

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary and to mitigate accidents.

The ESFAS contains devices and circuitry that generate the following signals when the monitored variables reach levels that are indicative of conditions requiring protective action:

1. Safety Injection Actuation Signal (SIAS),
2. Containment Spray Actuation Signal (CSAS),
3. Containment Isolation Actuation Signal (CIAS),
4. Main Steam Isolation Signal (MSIS),
5. Recirculation Actuation Signal (RAS), and
6. Auxiliary Feedwater Actuation Signal (AFAS).

Equipment actuated by each of the above signals is identified in the FSAR (Ref. 1).

Each of the above ESFAS actuation systems is segmented into four sensor subsystems addressed by LCO 3.3.~~4~~, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," and two actuation subsystems addressed by this LCO. Each sensor subsystem includes measurement channels and bistables. The SIAS actuation subsystems include two logic subsystems for sequentially loading the diesel generators.

Each of the four sensor subsystem channels monitors redundant and independent process measurement channels. Each sensor is monitored by at least one bistable. The bistable associated with each ESFAS Function will trip when the monitored variable exceeds the trip setpoint. When tripped, the sensor subsystems provide outputs to the two actuation subsystems.

BASES

BACKGROUND (continued)

The two independent actuation subsystems each compare the four associated sensor subsystem outputs. If a trip occurs in two or more sensor subsystem channels, the two-out-of-four logic in each actuation subsystem will initiate one train of ESFAS. Each has sufficient equipment to provide protection to the public in the case of a Design Basis Event. The sensor subsystem is addressed in LCO 3.3.~~4~~ 3. This LCO addresses the actuation subsystem. 3

Each of the four sensor subsystems is mounted in a separate cabinet, excluding the sensors and field wiring.

The role of the sensor subsystem (measurement channels and bistables) is discussed in LCO 3.3.~~4~~ 3. That of the actuation subsystem is discussed below. 3

ESFAS Logic

The two independent actuation subsystems compare the four sensor subsystem outputs. If a trip occurs in the same parameter in two or more sensor subsystem channels, the two-out-of-four logic in each actuation subsystem initiates one train of ESFAS. Either train controls sufficient redundant and independent equipment.

INSERT 1

Each actuation subsystem channel is housed in two cabinets. One cabinet contains the logic circuitry for the actuation channel, while the other cabinet contains the power relay equipment. This power relay equipment includes the power relays (initiation relays) that actuate the ESFAS equipment in response to a signal from the Actuation Logic.

It is possible to change the two-out-of-four ESFAS Logic to a two-out-of-three logic for a given input parameter in one channel at a time by disabling one channel input to the logic. Thus, the bistables will function normally, producing normal trip indication and annunciation, but ESFAS actuation will not occur since the bypassed channel is effectively removed from the coincidence logic. Maintenance bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. ~~At some plants an interlock prevents simultaneous maintenance bypassing of the same parameter in more than one channel.~~ Maintenance bypassing is normally employed during maintenance or testing. 1

1

INSERT 1

Each ESFAS consists of four measurement channels (designated MA, MB, MC, and MD) for each input parameter, two logic matrix systems (SA and SB) and two actuation channels (A and B).

Each measurement channel consists of a sensor, power supply and bistable unit arranged in a current flow loop circuit. The bistable unit provides a digital signal to logic matrices where signals from all four measurement channels for that parameter are combined in a two-out-of-four logic network. Isolation devices are provided to maintain separation between the measurement channels and the logic matrices. Logic matrices (SA or SB) provide initiation signals to their associated actuation channels (A or B) when the logic for the particular signal is satisfied.

BASES

BACKGROUND (continued)

~~For plants that have demonstrated sufficient channel to channel independence,~~ two-out-of-three logic is the minimum that is required to provide adequate plant protection, since a failure of one channel still ensures that ESFAS actuation would be generated by the two remaining OPERABLE channels. Two-out-of-three logic also prevents inadvertent actuation caused by any single channel failure in a trip condition. 1

In addition to the maintenance bypasses, there are operating bypasses (blocks) on the Pressurizer Pressure - Low input to the SIAS and on the Steam Generator Pressure - Low input to the MSIS when these inputs are no longer required for protection. These bypasses are enabled manually when the enabling conditions are satisfied in three of the four sensor subsystem channels. The operating bypass circuitry employs four bistable channels in the sensor subsystems, sensing pressurizer pressure (for the SIAS) and steam generator pressure (for the MSIS). These bistables provide contact output to the three-out-of-four logic in the two actuation subsystem channels. When the logic is satisfied, manual bypassing is permitted. There are two manual bypass actuation controls for each Function, one per train.

All operating bypasses are automatically removed when enabling bypass conditions are no longer satisfied.

Manual ESFAS initiation capability is provided to permit the operator to manually actuate an Engineered Safety Features (ESF) System when necessary. Two push buttons are provided in the control room for each ESFAS Function. Each push button actuates one train via the ESFAS Logic.

The Actuation Logic is tested by inserting a local test signal. A coincidence logic trip will occur if there is the simultaneous presence of a sensor channel trip, either legitimate or due to testing. Most ESFAS Functions employ several separate parallel two-out-of-four Actuation Logic modules, with each module actuating a subset of the ESFAS equipment associated with that Function. Each of these subchannels can be tested individually so that simultaneous actuation of an entire train can be avoided during testing.

INSERT 2

~~Except in the case of actuation subchannels SIAS Nos. 5 and 10, CIAS No. 5, and MSIS No. 1, all Actuation Logic channels can be tested at power. The above designated subchannels must be tested when shut down because they actuate the following equipment, which cannot be actuated at power:~~ 1

1

INSERT 2

Some logic channels are testable during reactor operation without affecting the operability or safety of the plant. These devices are tested by imposing an ESFAS during reactor operation. Other logic channels are not testable during reactor operation and must be tested when shutdown to prevent affecting the operability or safety of the plant. Testing Criteria are identified in the UFSAR (Ref. 1).

BASES

BACKGROUND (continued)

- ~~Reactor coolant pump (RCP) seal bleedoff isolation valves,~~
- ~~Service water isolation valves,~~
- ~~Volume control tank (VCT) discharge valves,~~
- ~~Letdown stop valves,~~
- ~~Component cooling water (CCW) to RCPs,~~
- ~~CCW from RCPs,~~
- ~~Main steam isolation valves (MSIVs),~~
- ~~Feedwater isolation valves, and~~
- ~~Instrument air containment isolation valves.~~

1

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. Functions such as Manual Initiation, not specifically credited in the accident analysis, serve as backups to Functions and are part of the NRC staff approved licensing basis for the plant.

ESFAS protective Functions are as follows:

1. Safety Injection Actuation Signal

   The SIAS ensures acceptable consequences during loss of coolant accident (LOCA) events, including steam generator tube rupture, and main steam line breaks (MSLBs) or feedwater line breaks (FWLBs) (inside containment). To provide the required protection, either a high containment pressure or a low pressurizer pressure signal will initiate SIAS. SIAS initiates the Emergency Core Cooling Systems (ECCS) and performs several other Functions, such as initiating control room isolation and starting the diesel generators.

BASES

APPLICABLE SAFETY ANALYSES (continued)

2. Containment Spray Actuation Signal

The CSAS initiates containment spray, preventing containment overpressurization during a LOCA or MSLB. ~~At some plants,~~ both a high containment pressure signal and an SIAS have to actuate to provide the required protection. This configuration reduces the likelihood of inadvertent containment spray. 1

3. Containment Isolation Actuation Signal

The CIAS actuates the Containment Isolation System, ensuring acceptable consequences during LOCAs and MSLBs or FWLBs (inside containment). To provide protection, a high containment pressure signal will initiate CIAS at the same setpoint at which an SIAS is initiated.

4. Main Steam Isolation Signal

The MSIS ensures acceptable consequences during an MSLB or FWLB by isolating both steam generators if either generator indicates a low steam generator pressure. The MSIS, concurrent with or following a reactor trip, minimizes the rate of heat extraction and subsequent cooldown of the RCS during these events.

5. Recirculation Actuation Signal

At the end of the injection phase of a LOCA, the refueling water tank (RWT) will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from RWT to containment sump must occur before the RWT empties to prevent damage to the ECCS pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support pump suction. Furthermore, early switchover must not occur to ensure sufficient borated water is injected from the RWT to ensure the reactor remains shut down in the recirculation mode. An RWT Level - Low signal initiates the RAS.

BASES

APPLICABLE SAFETY ANALYSES (continued)

6. Auxiliary Feedwater Actuation Signal

the affected

An AFAS initiates feedwater flow to ~~both~~ steam generator~~s~~ if a low level is indicated in ~~either~~ steam generator, unless the generator is ~~ruptured~~.

the affected

faulted

The AFAS maintains a steam generator heat sink during the following events:

- MSLB,
- FWLB,
- Inadvertent opening of a steam generator atmospheric dump valve, and
- Loss of feedwater.

A low steam generator water level signal will initiate auxiliary feed to the affected steam generator.

INSERT 3

1

Secondary steam generator (SG) differential pressure (SG-A > SG-B) or (SG-B > SG-A) inhibits auxiliary feed to a generator identified as being ~~ruptured~~. This input to the AFAS logic prevents loss of the intact generator while preventing feeding a ~~ruptured~~ generator during MSLBs and FWLBs. This prevents containment overpressurization during these events.

faulted

faulted

INSERT 4

1

The ESFAS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that all components necessary to provide an ESFAS actuation be OPERABLE.

Actions allow maintenance bypass of individual channels. Plants are restricted to 48 hours in a maintenance bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic).

The Bases for the LCO on ESFAS automatic actuation Functions are addressed in the Bases for LCO 3.3.~~4~~. Those associated with the Manual ~~Trip~~ or Actuation Logic are addressed below.

3

Actuation

3

1

INSERT 3

A separate auxiliary feedwater actuation signal is generated for each steam generator (AFAS-1, AFAS-2). For each AFAS-1 and AFAS-2 there are four independent level transmitters for steam generator level, four independent steam generator pressure transmitters, and four independent feedwater header pressure transmitters. The AFAS actuation logic actuates auxiliary feedwater to a steam generator on low level after a time delay period unless that steam generator or its associated auxiliary feedwater supply header have been identified as being faulted. R1

1

INSERT 4

The AFAS actuation logic isolates auxiliary feedwater flow to a steam generator upon recovery of steam generator level.

BASES

LCO (continued)

1. Safety Injection Actuation Signal

   a. Manual ~~Trip~~

   This LCO requires two channels of SIAS Manual ~~Trip~~ to be OPERABLE in MODES 1, 2, 3, and 4.

   b. Actuation Logic

   This LCO requires two channels of SIAS Actuation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

   ~~Failures in the actuation subsystems, including the manual bypass key switches, are Actuation Logic failures and are addressed in this LCO.~~

   Actuation Logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

2. Containment Spray Actuation Signal

   CSAS is initiated either manually or automatically. ~~At many plants~~ it is also necessary to have an automatic or manual SIAS for a complete actuation. ~~The SIAS opens the containment spray valves, whereas the CSAS actuates other required components~~. The SIAS requirement should always be satisfied on a legitimate CSAS, since the Containment Pressure - High signal used in the SIAS ~~is the same~~ setpoint used in the CSAS. ~~The transmitters used to initiate CSAS are independent of those used in the SIAS to prevent inadvertent containment spray due to failures in two sensor channels.~~

   a. Manual ~~Trip~~

   This LCO requires two channels of CSAS Manual ~~Trip~~ to be OPERABLE in MODES 1, 2, 3, and 4.

   b. Actuation Logic

   This LCO requires two channels of CSAS Actuation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

   Actuation Logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

BASES

LCO (continued)

3. Containment Isolation Actuation Signal

a. Manual ~~Trip~~ (Actuation) 1

This LCO requires two channels of CIAS Manual ~~Trip~~ (Actuation) to be OPERABLE in MODES 1, 2, 3, and 4. 1

b. Actuation Logic

This LCO requires two channels of Actuation Logic for CIAS to be OPERABLE in MODES 1, 2, 3, and 4.

Actuation Logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

4. Main Steam Isolation Signal

a. Manual ~~Trip~~ (Actuation) 1

This LCO requires two channels ~~per steam generator~~ of the MSIS Manual ~~Trip~~ (Actuation) to be OPERABLE in MODES 1, 2, 3~~, and 4~~ (and except when all MSIVs are closed). 1

Each channel actuates its associated MSIS Actuation Logic channel. Actuating either MSIS Manual Actuation will result in a full main steam isolation.

b. Actuation Logic 1

This LCO requires two channels of MSIS Actuation Logic to be OPERABLE in MODE~~S~~ 1, 2~~,~~ 3~~, and 4~~ (and MODES and except when all MSIVs are closed). 1

Each MSIS Actuation Logic channel provides a signal to its respective MSIV. An interposing relay will cause the opposite train MSIV to close.

~~Failures in the actuation subsystems, including the manual bypass key switches, are considered Actuation Logic failures and are addressed in the logic LCO.~~

5. Recirculation Actuation Signal

a. Manual ~~Trip~~ (Actuation) 1

This LCO requires two channels of RAS Manual ~~Trip~~ (Actuation) to be OPERABLE in MODES 1, 2, 3, and 4. 1

b. Actuation Logic

This LCO requires two channels of RAS Actuation Logic to be OPERABLE in MODES 1, 2, 3, and 4.

BASES

LCO (continued)

6. Auxiliary Feedwater Actuation Signal

the affected steam
steam
its
the affected
faulted
Actuation
Actuation

A low level in ~~either~~ generator, as sensed by a two-out-of-four coincidence of four wide range sensors for each generator, will generate an auxiliary feedwater actuation signal (AFAS), which starts ~~both~~ train~~s~~ of auxiliary feedwater (AFW) pumps and feeds ~~both~~ steam generator~~s~~. The AFAS also monitors the secondary differential pressure in both steam generators and initiates an AFAS block signal to a ~~ruptured~~ generator if the pressure in that generator is lower than the other generator by the differential pressure setpoint.

1
1
1
1
1

a. Manual ~~Trip~~

This LCO requires two channels of AFAS Manual ~~Trip~~ to be OPERABLE in MODES 1, 2, and 3.

b. Actuation Logic

This LCO requires two channels of AFAS Actuation Logic to be OPERABLE in MODES 1, 2, and 3.

Actuation Logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

APPLICABILITY All ESFAS Functions are required to be OPERABLE in MODES 1, 2, and 3. In MODES 1, 2, and 3, there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the MSIVs to preclude a positive reactivity addition,
- Actuate AFW to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available),
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or MSLB, and
- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

BASES

APPLICABILITY (continued)

In MODES 4, 5, and 6, automatic actuation of ESFAS Functions is not required, because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating the ESF components if required. ESFAS Manual ~~Trip~~ capability is required for Functions other than AFAS and MSIS in MODE 4 even though automatic actuation is not required. Because of the large number of components actuated on each ESFAS, actuation is simplified by the use of the Manual ~~Trip~~ push buttons. Manual ~~Trip~~ of AFAS is not required in MODE 4 because AFW or shutdown cooling will already be in operation in this MODE.

The ESFAS Actuation Logic must be OPERABLE in the same MODES as the Automatic and Manual ~~Trips~~. In MODE 4, only the portion of the ESFAS logic responsible for the required Manual ~~Trip~~ must be OPERABLE.

In MODES 5 and 6, ESFAS initiated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components.

ACTIONS

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered, if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.~~5~~-1 in the LCO. Completion Times for the inoperable channel of a Function will be tracked separately.

A.1

Condition A applies to one AFAS Manual ~~Trip~~ or AFAS Actuation Logic channel inoperable. It is ~~identical~~ to Condition ~~C~~ for the other ESFAS Functions, except for the shutdown track imposed by Condition ~~D~~.

The channel must be restored to OPERABLE status to restore redundancy of the AFAS Function. The 48 hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~

Actuation

4

3

BASES

ACTIONS (continued)

B.1 and B.2

Actuation

If two Manual ~~Trip~~ or Actuation Logic channels are inoperable or the Required Action and associated Completion Time of Condition A cannot be met, the reactor should be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within ~~[~~12~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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E

~~C~~.1

E

F

Actuation

and MSIS

3

Condition ~~C~~ applies to one Manual ~~Trip~~ or Actuation Logic channel inoperable for those ESFAS Functions that must be OPERABLE in MODES 1, 2, 3, and 4 (all Functions except AFAS). The shutdown track imposed by Condition ~~D~~ requires entry into MODE 5, where the LCO does not apply to the affected Functions. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~

1

2

The channel must be restored to OPERABLE status to restore redundancy of the affected Functions. The 48 hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel.

~~D.1 and D.2~~

1

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

2

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

4

3

1

St. Lucie – Unit 2

Revision XXX

1

INSERT 5

C.1

Condition C applies when one MSIS Manual Actuation or Actuation Logic channel is inoperable.

The 48 hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the remaining OPERABLE MSIS Manual Actuation or Actuation Logic channel. Alternatively, the Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

D.1.1, D.1.2, D.2.1, and D.2.2

If both MSIS Manual Actuation or Actuation Logic channels are inoperable or if the Required Actions and associated Conditions of Condition C are not met, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by closing the MSIVs and MFIVs within 6 hours (Required Actions D.1.1 and D.1.2). If allowed, operation in MODES 2 or 3 with MSIVs and MFIVs closed may continue because isolating the affected main steam line and main feedwater line accomplishes the safety function of the inoperable channel. Alternately, if the MSIVs and MFIVs cannot be closed, the plant must be placed in MODE 3 within 6 hours and in MODE 4 within 12 hours (Required Actions D.2.1 and D.2.2).

The Completion Times of Required Actions D.1.1 and D.1.2 is consistent with the Completion Time to be in MODE 3 for similar conditions and correspond to the Completion Time of D.2.1. The Completion Times of Required Actions D.2.1 and D.2.2 are based on the Completion Time to reach MODE 3 (i.e., 6 hours) and allow an additional 6 hours to reach MODE 4, respectively, if the MSIVs and MFIVs are not closed within 12 hours. These Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

~~Condition D is entered when the Required Action and associated Completion Time of Condition C is not met. The plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.~~ 1

~~Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

~~Required Action D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~

F
F
F
~~E~~.1 and ~~E~~.2

Condition ~~E~~ is entered when one or more Functions have two Manual ~~Trip~~ or Actuation Logic channels inoperable except AFAS. The plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

and MSIS
Actuation
3
1

SURVEILLANCE REQUIREMENTS

SR 3.3.~~5~~.1
4

A CHANNEL FUNCTIONAL TEST is performed to ensure the entire channel will perform its intended function when needed. Sensor subsystem tests are addressed in LCO 3.3.~~4~~. This SR addresses Actuation Logic tests. A successful test of the required contact(s) of a

3

BASES

SURVEILLANCE REQUIREMENTS (continued)

channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Actuation Logic Tests

Actuation subsystem testing includes injecting one trip signal into each two-out-of-four logic subsystem in each ESFAS Function and using a bistable trip input to satisfy the trip logic. Initiation relays associated with the affected channel will then actuate the individual ESFAS components. Since each ESFAS Function employs subchannels of Actuation Logic, it is possible to actuate individual components without actuating an entire ESFAS Function.

Note 1 requires that Actuation Logic tests include operation of initiation relays. Note 2 allows deferred at power testing of certain relays to allow for the fact that operating certain relays during power operation could cause plant transients or equipment damage. ~~Those initiation relays that cannot be tested at power must be tested in accordance with Note 2. These include [SIAS No. 5, SIAS No. 10, CIAS No. 5, and MSIS No. 1.]~~ 1 2

~~These relays actuate the following components, which cannot be tested at power:~~

- ~~RCP seal bleedoff isolation valves,~~
- ~~Service water isolation valves,~~ 1
- ~~VCT discharge valves,~~
- ~~Letdown stop valves,~~
- ~~CCW to and from the RCPs,~~
- ~~MSIVs and feedwater isolation valves, and~~
- ~~Instrument air containment isolation valves.~~

~~The reasons that each of the above cannot be fully tested at power are stated in Reference 1.~~

These tests verify that the ESFAS is capable of performing its intended function, from bistable input through the actuated components.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[The Frequency of [92] days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 3).~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.3.~~5~~.2

A CHANNEL FUNCTIONAL TEST is performed on the manual ESFAS actuation circuitry, de-energizing relays and providing Manual ~~Trip~~ of the Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

This Surveillance verifies that the trip push buttons are capable of opening contacts in the Actuation Logic as designed, de-energizing the initiation relays and providing Manual ~~Trip~~ of the Function. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at a Frequency of once every [18] months.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

REFERENCES

1. FSAR, Section ~~[~~7.3~~]~~. U
2. ~~CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.~~
3. ~~CEN-327, June 2, 1986, including Supplement 1, March 3, 1989.~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.4 BASES, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) LOGIC AND MANUAL ACTUATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The heading for ISTS 3.3.5 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2). Therefore, ISTS 3.3.2 is not included in the PSL ITS. The successive Specifications are renumbered as necessary.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.4, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS)
LOGIC AND MANUAL ACTUATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L01

Florida Power & Light Company (FPL) is converting the St. Lucie Plant (PSL) Unit 1 and Unit 2 Technical Specifications to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 5. The proposed change involves making the current technical specifications (CTS) less restrictive. Below is the description of the change and the determination of No Significant Hazards Considerations for conversion to NUREG-1432.

Unit 1 CTS 3.3.2.1, Table 3.3-3 Functional Unit 4 – main steam isolation signal (MSIS) manual actuation is applicable in MODES 1, 2, 3, and 4 and Table 4.3-2 Functional Unit 4 – MSIS automatic actuation logic is applicable in MODES 1, 2, and 3. Unit 2 CTS 3.3.2, Tables 3.3-3 and 4.3-2 Functional Unit 4 – MSIS manual and automatic actuation logic are applicable in MODES 1, 2, and 3. When one Unit 1 MSIS manual actuation channel is inoperable or one Unit 2 MSIS automatic actuation logic channel is inoperable, CTS Table 3.3-3 Action 8 (Unit 1) and Action 12 (Unit 2) require, in part, to restore the channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program.

ITS Limiting Condition for Operation (LCO) 3.3.4 (Table 3.3.4-1) requires the MSIS manual actuation and automatic actuation channel to be OPERABLE in MODES 1, 2, 3 except in MODES 2 and 3 when valves isolated by the MSIS Function are closed as specified in Footnote (a). ITS actions require closing the main steam isolation valves (MSIVs) and the main feedwater isolation valves (MFIVs) in 6 hours, or be in MODE 3 in 6 hours and MODE 4 in 12 hours when two MSIS Manual Actuation or Actuation Logic channels are inoperable or when the action to restore one MSIS Manual Actuation or Actuation Logic channel cannot be completed within the required Completion Time. This changes the CTS by deleting the MODE 4 applicability requirement for Unit 1 MSIS, and changing the MODE 2 and 3 Applicability for both units to include “except when valves isolated by the MSIS Function are closed.” This change aligns the Applicability of the MSIS actuation instrumentation with the applicability requirements of the associated MSIS instrument channels (i.e., steam generator low pressure instrument channels and the Unit 2 containment high pressure instrument channels) and the supported equipment (i.e., the MSIVs and MFIVs). The change also revises the CTS Actions to add appropriate actions to reflect the change to the Applicability, which provides sufficient remedial actions to safely continue operation in lieu of a plant cooldown.

The purpose of the MSIS manual and automatic instrumentation in CTS Table 3.3-3 is to ensure the Engineered Safety Features Actuation System (ESFAS) MSIS Functions are OPERABLE when the supported equipment is required to be OPERABLE. Automatic steam line and feedwater line isolation is assumed in the mitigation of a major secondary system pipe rupture accident (e.g., main steam line break or feedwater line break). The MSIS manual actuation instrumentation is not credited in the safety analysis and a safety analysis limit is not specified for this actuation function. MSIS automatic actuation instrumentation is credited in the safety analysis.

In MODES 1, 2, and 3, there is sufficient energy in the primary and secondary systems to warrant automatic ESF System response to ensure the main steam and feedwater

lines can be isolated in the event of a high energy secondary system pipe rupture. The exception to MODE 2 and 3 are added to clarify that the MSIS manual and automatic actuation instrumentation is not required to be OPERABLE when the valves actuated by the MSIS instrumentation are in positions that support the safety analyses. When the valves isolated by the MSIS function are in the closed position, they are in their assumed accident position. This change is acceptable, because when the MSIVs and MFIVs are closed, the adverse effects of a high energy secondary system pipe rupture are precluded and the requirements continue to ensure that the structures, systems, and components (SSCs) are maintained in the MODES and other specified conditions assumed in the safety analyses. The proposed applicability and actions provide the same level of protection as the current requirements for the supported equipment.

Since the proposed actions provide the allowance to place the MSIVs and MFIVs in their accident position with the option to continue a plant cooldown, sufficient remedial measures continue to be provided to allow safe operation pursuant to the requirements of 10 CFR 50.36(c)(2).

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No.

   The proposed change does not affect accident initiators or precursors nor adversely alter the design assumptions, conditions, and configuration of the facility. The proposed change does not alter any plant equipment or operating practices with respect to such initiators or precursors in a manner that the probability of an accident is increased. The proposed change revises the applicability requirements for the ESFAS MSIS actuation instrumentation to align with the applicability requirements of the associated instrument channels and the supported equipment. Changing the applicability requirements of the MSIS actuation instrumentation to be consistent with the applicability requirements of the associated instrument channels and the supported equipment does not alter the probability or consequences of previously evaluated accidents. In addition, the proposed actions provide sufficient remedial measures to continue safe operation in lieu of a plant cooldown. The proposed change does not involve a physical change to the ESFAS, nor does it change the safety function of the ESFAS instrumentation or the equipment supported by the ESFAS instrumentation. Automatic steam line and feedwater line isolation is assumed in the mitigation of a major secondary system pipe rupture accident. When the valves are in the closed position, they are in their assumed accident position. Therefore, the adverse effects of a high energy secondary system pipe rupture are precluded. As a result, the proposed change does not alter assumptions relative to the mitigation of a previously evaluated accident.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No

   With respect to a new or different kind of accident, there are no proposed design changes to the ESFAS; nor are there any changes in the method by which safety related plant SSCs perform their specified safety functions. The proposed change will not affect the normal method of plant operation or revise any operating parameters. No new accident scenarios, transient precursor, failure mechanisms, or limiting single failures will be introduced as a result of this proposed change and the failure modes and effects analyses of SSCs important to safety are not altered as a result of this change.

   The proposed change does not alter the design or performance of the ESFAS, rather, the change aligns the applicability requirements of the MSIS actuation instrumentation with the applicability requirements of the associated instrument channels and the supported equipment. The proposed actions provide sufficient remedial measures to continue safe operation in lieu of a plant cooldown. The process to close the MSIVs and the MFIVs or place the unit in a shutdown condition uses current procedures, methods, and processes already established and currently in use and, therefore, does not constitute a new type of test.

   No changes are being proposed to the procedures that operate the plant equipment and the change does not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal.

   Therefore, the proposed change will not create the possibility of a new or different accident previously evaluated.

3. Does the proposed change involve a significant reduction in the margin of safety?

   Response: No.

   The margin of safety is related to the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and the containment. The performance of these fission product barriers is not affected by the proposed change.

   Instrumentation safety margin is established by ensuring the limiting safety system settings (LSSSs) automatically actuate the applicable design function to correct an abnormal situation before a safety limit is exceeded. Safety analysis limits are established for reactor trip system and ESFAS instrumentation functions related to those variables having significant safety functions. The proposed change aligns the applicability requirements of the MSIS actuation instrumentation with the applicability requirements of the associated instrument channels and the supported equipment. The proposed actions provide sufficient remedial measures to continue safe operation in lieu of a plant cooldown. The MSIS manual actuation instrumentation is not credited in the safety analysis and a safety analysis limit is not specified for this actuation function. Therefore, the MSIS manual actuation instrumentation does not

represent an LSSS because this instrumentation does not monitor a plant variable on which a safety limit has been placed. In addition, the change to the applicability and actions of the MSIS automatic actuation logic does not alter the capability of the LSSSs to automatically actuate the applicable design function to correct an abnormal situation before a safety limit is exceeded because the applicability and optional remedial actions to close the associated valves ensure the components are in their safety position precluding adverse effects of a high energy secondary system pipe rupture. The proposed change does not alter the instrumentation used to monitor a plant variable on which a safety limit has been placed.

The controlling parameters established to isolate the steam and feedwater lines during an accident or transient are not affected by the proposed change and no design basis or safety limit is altered as a result of the proposed change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above analysis, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of no significant hazards consideration is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.4, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS)
LOGIC AND MANUAL ACTUATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L04 – UNIT 2
LESS RESTRICTIVE CHANGE L05 – UNIT 1

Florida Power & Light Company (FPL) is converting the St. Lucie Plant (PSL) Unit 1 and Unit 2 Technical Specifications to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 5. The proposed change involves making the current technical specifications (CTS) less restrictive. Below is the description of the change and the determination of No Significant Hazards Considerations for conversion to NUREG-1432.

When one Unit 2 main steam isolation signal (MSIS) manual trip button channel is inoperable, CTS actions require the channel to be restored to OPERABLE status within 48 hours or declare the associated valve inoperable and take the actions required by the supported system specification. ITS requires restoring the channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time (RICT) Program. This changes the CTS by optionally allowing the manual channel to be restored in accordance with the RICT Program provided the associated risk is assessed and managed and in accordance with an NRC approved RICT Program. This change aligns the action for one inoperable Unit 2 MSIS manual channel with the action for one inoperable Unit 1 MSIS manual channel and the action for one inoperable channel of the other Engineered Safety Features Actuation System (ESFAS) manual actuation channels.

Unit 1 CTS does not specifically include requirements for the ESFAS automatic actuation logic channels, including actions when one or more automatic actuation logic channels are inoperable. However, Surveillance Requirements associated with the ESFAS automatic actuation logic is included in the Unit 1 CTS. Because no actions are provided for the Unit 1 ESFAS automatic actuation logic channels, a plant shutdown is required when an automatic actuation channel is inoperable. When an ESFAS automatic actuation logic channel is inoperable, Unit 1 ITS actions allow restoring a channel to OPERABLE status within 48 hours or in accordance with the RICT Program consistent with the Unit 2 ITS actions. This changes the Unit 1 CTS by providing time to restore the redundant automatic actuation logic channel to OPERABLE status provided the associated risk is assessed and managed in accordance with an NRC approved RICT Program.

License Amendments 247 and 199, dated July 2, 2019, for PSL Unit 1 and Unit 2, respectively (NRC ADAMS Accession No. ML19113A099) incorporated an NRC approved RICT Program in the PSL Unit 1 and Unit 2 technical specifications. The RICT Program provides controls to assess and manage risk during the extended time to restore the redundant automatic actuation logic channel or manual actuation channel to OPERABLE status.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.4, ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) LOGIC AND MANUAL ACTUATION

1. Does the proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No.

   The proposed change does not affect accident initiators or precursors nor adversely alter the design assumptions, conditions, and configuration of the facility. The proposed change does not alter any plant equipment or operating practices with respect to such initiators or precursors in a manner that the probability of an accident is increased. In addition, the actions continue to provide sufficient remedial measures to continue safe operation in lieu of a plant shutdown. The proposed change does not involve a physical change to the ESFAS, nor does it change the safety function of the ESFAS instrumentation or the equipment supported by the ESFAS instrumentation.

   The proposed change permits the extension of the time to restore the redundant automatic actuation logic channel or manual actuation channel to OPERABLE status provided the associated risk is assessed and managed in accordance with an NRC approved RICT Program. The proposed change does not involve a significant increase in the probability of an accident previously evaluated because the change involves no change to the plant or its modes of operation. The proposed change does not increase the consequences of an accident because the design basis mitigation function of the ESFAS instrumentation and supported systems is not changed and the consequences of an accident during the extended Completion Time are no different from those during the existing Completion Time

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No

   With respect to a new or different kind of accident, there are no proposed design changes to the ESFAS; nor are there any changes in the method by which safety related plant structures systems and components perform their specified safety functions. The proposed change will not affect the method of plant operation or revise any operating parameters. No new accident scenarios, transient precursor, failure mechanisms, or limiting single failures will be introduced as a result of this proposed change and the failure modes and effects analyses of SSCs important to safety are not altered as a result of this change.

   The proposed change does not alter the design or performance of the ESFAS, rather, the change permits the extension of the time to restore the redundant automatic actuation logic channel or manual actuation channel to OPERABLE status provided the associated risk is assessed and managed in accordance with an NRC approved RICT Program. The proposed actions provide sufficient remedial measures to continue safe operation in lieu of a plant shutdown. The process to return inoperable instrumentation to service or place the unit in a shutdown condition

uses current procedures, methods, and processes already established and currently in use and, therefore, does not constitute a new type of test.

No changes are being proposed to the procedures that operate the plant equipment and the change does not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal.

Therefore, the proposed change will not create the possibility of a new or different accident previously evaluated.

3. Does the proposed change involve a significant reduction in the margin of safety?

   Response: No.

   The margin of safety is related to the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and the containment. The performance of these fission product barriers is not affected by the proposed change.

   Instrumentation safety margin is established by ensuring the limiting safety system settings (LSSSs) automatically actuate the applicable design function to correct an abnormal situation before a safety limit is exceeded. Safety analysis limits are established for reactor trip system and ESFAS instrumentation functions related to those variables having significant safety functions. The proposed actions provide sufficient remedial measures to continue safe operation in lieu of a plant shutdown. In addition, the change does not allow continued operation with a loss of the actuation capability, and therefore, does not alter the capability of the LSSSs to automatically actuate the applicable design function to correct an abnormal situation before a safety limit is exceeded.

   The proposed change does not alter the instrumentation used to monitor a plant variable on which a safety limit has been placed. The change permits the extension of the time to restore the redundant automatic actuation logic channel or manual actuation channel to OPERABLE status provided the associated risk is assessed and managed in accordance with an NRC approved RICT Program.

   The controlling parameters established to actuate supporting systems during an accident or transient are not affected by the proposed change and no design basis or safety limit is altered as a result of the proposed change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above analysis, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of no significant hazards consideration is justified.

ATTACHMENT 5

3.3.5, Diesel Generator (DG) – Loss of Voltage Start (LOVS)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.5

~~3.3.2.1~~ The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE ~~with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4~~.

See ITS 3.3.3

LA01

LCO 3.3.5 Applicability

APPLICABILITY: As shown in Table 3.3-3.

Add proposed ITS 3.3.5 Applicability: When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

M01

Add proposed ITS 3.3.5 ACTIONS Note

A02

ACTION:

SR 3.3.5.3

ACTION A for Table 3.3.3-1 Functions 6.a, 6.b, 6.c

SR 3.3.5.3

~~a.~~ With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, ~~declare the channel inoperable and~~ apply the applicable ACTION requirement of Table 3.3-3 ~~until the channel is restored to OPERABLE status~~ with the trip setpoint adjusted consistent with the Trip Setpoint value.

ACTION A for Table 3.3.3-1 Functions 6.a, 6.b, 6.c

~~b.~~ With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

Add proposed ITS 3.3.5 ACTION B and C

L01

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1, SR 3.3.5.2, SR 3.3.5.3

~~4.3.2.1.1~~ Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes ~~and at the frequencies~~ shown in Table 4.3-2.

In accordance with the Surveillance Frequency Control Program

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

See ITS 3.3.3

~~4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per function.~~

A03

A01

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LA02

A04

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ REQUIRED CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	13
6. LOSS OF POWER					
a. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	~~2/Bus~~	~~2/Bus~~	2 → ~~1~~/Bus	1, 2, 3	12
b. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)	~~2/Bus~~	~~2/Bus~~	2 → ~~1~~/Bus	1, 2, 3	12
v c. 480 V Emergency Bus Under-voltage (Degraded Voltage)	~~2/Bus~~	~~2/Bus~~	2 → ~~1~~/Bus	1, 2, 3	12
7. AUXILIARY FEEDWATER (AFAS)					
a. Manual (Trip Buttons)	4/SG	2/SG	4/SG	1, 2, 3	11
b. Automatic Actuation Logic	4/SG	2/SG	3/SG	1, 2, 3	11
c. SG Level (1A/1B) - Low	4/SG	2/SG	3/SG	1, 2, 3	14a, 14b, 15
8. AUXILIARY FEEDWATER ISOLATION					
a. SG 1A – SG 1B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	14a, 14b, 15
b. Feedwater Header 1A – 1B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	14a, 15

See ITS 3.3.4 ITS 3.3.3

See ITS 3.3.4 ITS 3.3.3

See ITS 3.3.4 ITS 3.3.3

M01

Add proposed ITS 3.3.5 Applicability:
When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

LCO 3.3.5 Loss of Voltage ACTION A

LCO 3.3.5 Degraded Voltage ACTION A

LCO 3.3.5 Degraded Voltage ACTION A

Add proposed ITS 3.3.5 ACTION B and C

L01

A01

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION 10A - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the bypassed or tripped condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then place the inoperable channel in the tripped condition.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

ACTION 10B - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

See ITS 3.3.3

ACTION 11 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.4

ACTION 12 - ~~With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable~~ channel ~~is~~ placed in ~~the~~ trip~~ped~~ ~~condition~~ within 1 hour.

L02

ACTION A

ACTION A (Condition)

One or more Functions with one channel per DG inoperable.

Add proposed ITS 3.3.5 ACTIONS Note

A02

Add proposed ITS 3.3.5 ACTION B and C

L01

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

LA01

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES	
6. LOSS OF POWER			
SR 3.3.5.3 a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	~~≥ 2900 volts with a 1 ± .5 second time delay~~	≥ 2900 volts with a 1 ± .5 second time delay	
SR 3.3.5.3 b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	~~≥ 3831 volts with a 18 ± 2 second time delay~~	≥ 3831 volts with a 18 ± 2 second time delay	
SR 3.3.5.3 c. 480 volts Emergency Bus Undervoltage (Degraded Voltage)	~~≥ 415 volts with a ≤ 9 second time delay~~	≥ 415 volts with a ≤ 9 second time delay	
7. AUXILIARY FEEDWATER (AFAS)			See ITS 3.3.4
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	
b. Automatic Actuation Logic	Not Applicable	Not Applicable	
Table 3.3.3-1 6.a. 6.b c. SG 1A & 1B Level Low	≥ 19.0%	≥ 18.0 %	
8. AUXILIARY FEEDWATER ISOLATION			See ITS 3.3.3
a. Steam Generator ΔP – High	≤ 275 psid	89.2 to 281 psid	
b. Feedwater Header High ΔP	≤ 150.0 psid	56.0 to 157.5 psid	

Add proposed ITS 3.3.5 Applicability:
When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

M01

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED	
6. LOSS OF POWER					
SR 3.3.5.3 a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	SFCP	SFCP	SFCP	1, 2, 3	
SR 3.3.5.3 b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3	
SR 3.3.5.3 c. 480 V Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3	
7. AUXILIARY FEEDWATER (AFAS)					
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	1, 2, 3	See ITS 3.3.4
b. SG Level (A/B) – Low	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP	1, 2, 3	See ITS 3.3.4
8. AUXILIARY FEEDWATER ISOLATION					
a. SG Level (A/B) – Low and SG Differential Pressure (BtoA/AtoB) – High	N.A.	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
b. SG Level (A/B) – Low and Feedwater Header Differential Pressure (BtoA/AtoB) – High	N.A.	SFCP	SFCP	1, 2, 3	

A01

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.5

~~3.3.2~~ The Engineered Safety Features Actuation System (ESFAS) instrumentation channels bypasses shown in Table 3.3-3 shall be OPERABLE ~~with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4~~.

See ITS 3.3.3

LA01

LCO 3.3.5 Applicability

APPLICABILITY: As shown in Table 3.3-3.

Add proposed ITS 3.3.5 Applicability: When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

M01

Add proposed ITS 3.3.5 ACTIONS Note

A02

ACTION:

SR 3.3.6.3

ACTION A for Table 3.3.3-1 Function 6.a.(1)
ACTION B for Table 3.3.3-1 Function 6.a.(2), 6.b.(1), 6.b.(2)
SR 3.3.6.3

~~a.~~ With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, ~~declare the channel inoperable and~~ apply the applicable ACTION requirement of Table 3.3-3 ~~until the channel is restored to OPERABLE status~~ with the trip setpoint adjusted consistent with the Trip Setpoint value.

ACTION A for Table 3.3.3-1 Function 6.a.(1)
ACTION B for Table 3.3.3-1 Function 6.a.(2), 6.b.(1), 6.b.(2)

~~b.~~ With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

Add proposed ITS 3.3.5 ACTION C and D

L01

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1, SR 3.3.5.2, SR 3.3.5.3

~~4.3.2.1~~ Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES ~~and at the frequencies~~ shown in Table 4.3-2.

In accordance with the Surveillance Frequency Control Program

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

See ITS 3.3.3

~~4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per function.~~

A03

A01

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LA02

A04

M01

FUNCTIONAL UNIT	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ CHANNELS OPERABLE	REQUIRED APPLICABLE MODES	ACTION
6. LOSS OF POWER (LOV)					
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	~~2/Bus~~	~~2/Bus~~	2 → ~~1~~/Bus	1, 2, 3	17A
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	~~3/Bus~~	~~2/Bus~~	3 → ~~2~~/Bus	1, 2, 3	17B
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	~~3/Bus~~	~~2/Bus~~	3 → ~~2~~/Bus	1, 2, 3	17B
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	~~3/Bus~~	~~2/Bus~~	3 → ~~2~~/Bus	1, 2, 3	17B
7. AUXILIARY FEEDWATER (AFAS)					
a. Manual (Trip Buttons)	4/SG	2/SG	4/SG	1, 2, 3	15
b. Automatic Actuation Logic	4/SG	2/SG	3/SG	1, 2, 3	15
c. SG Level (2A/2B) – Low	4/SG	2/SG	3/SG	1, 2, 3	20a, 20b, 21
8. AUXILIARY FEEDWATER ISOLATION					
a. SG 2A – SG 2B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	20a, 20b, 21
b. Feedwater Header 2A – 2B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	20a, 21

Add proposed ITS 3.3.5 Applicability: When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

LCO 3.3.5 Loss of Voltage ACTION A

LCO 3.3.5 Loss of Voltage ACTION B

LCO 3.3.5 Degraded Voltage ACTION B

LCO 3.3.5 Degraded Voltage ACTION B

See ITS 3.3.4

See ITS 3.3.3

See ITS 3.3.3

Add proposed ITS 3.3.5 ACTION C and D

L01

A01

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

See ITS 3.3.3

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below.

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Containment Pressure -	Containment Pressure – High (SIAS, CIAS, CSAS) Containment Pressure – High (RPS)
2. Steam Generator Pressure -	Steam Generator Pressure – Low (MSIS) AFAS-1 and AFAS-2 (AFAS) Thermal Margin/Low Pressure (RPS) Steam Generator Pressure – Low (RPS)
3. Steam Generator Level -	Steam Generator Level – Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)
4. Pressurizer Pressure -	Pressurizer Pressure – High (RPS) Pressurizer Pressure – Low (SIAS) Thermal Margin/Low Pressure (RPS)

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.4

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 17A - ~~With the number of OPERABLE Channels one less than the Total Number of Channels,~~ restore the inoperable channel to OPERABLE status within 48 hours or place ~~the inoperable~~ channel in ~~the~~ tripped ~~condition and verify that the Minimum Channels OPERABLE requirement is demonstrated~~ within ~~1~~ hour; one ~~additional~~ channel may be bypassed for up to 2 hours for surveillance testing ~~per Specification 4.3.2.1.~~

ACTION A

ACTION A Note

M02

48

s

One 4.16 kV Loss of Voltage Function with one channel per DG inoperable.

Add proposed ITS 3.3.5 ACTIONS Note

A02

Add proposed ITS 3.3.5 ACTIONS C and D

L01

A01

TABLE 3.3-3 (Continued)

One or more Functions with one channel per DG inoperable (except 4.16 kV Loss of Voltage Function).

TABLE NOTATION

ACTION B

ACTION 17B - ~~With the number of OPERABLE Channels one less than the Total Number of Channels,~~ restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or place ~~the inoperable~~ channel in ~~the~~ trip~~ped~~ ~~condition and verify that the Minimum Channels OPERABLE requirement is demonstrated~~ within ~~1~~ hour~~;~~ one ~~additional~~ channel may be bypassed for up to 2 hours for surveillance testing ~~per Specification 4.3.2.1~~.

ACTION B Note

48

s

M02

ACTION 18A - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

See ITS 3.3.3

a. The inoperable channel is placed in either the bypassed or tripped condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then place the inoperable channel in the tripped condition.

b. With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.

ACTION 18B - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 19 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. Within 1 hour the inoperable channel is placed in either the bypassed or tripped condition. If OPERABILITY cannot be restored within 48 hours or in accordance with the Risk Informed Completion Time Program, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

LA01

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES	
5. CONTAINMENT SUMP RECIRCULATION (RAS)			
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
b. Refueling Water Tank – Low	5.67 feet above tank bottom	4.62 feet to 6.24 feet above tank bottom	See ITS 3.3.3
c. Automatic Actuation Logic	Not Applicable	Not Applicable	See ITS 3.3.4
6. LOSS OF POWER			
SR 3.3.5.3 a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	~~≥ 3120 volts~~	≥ 3120 volts	
SR 3.3.5.3 (2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	~~≥ 360 volts~~	≥ 360 volts	
SR 3.3.5.3 b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	~~≥ 3848 volts with < 10-second time delay~~	≥ 3848 volts with < 10-second time delay	
SR 3.3.5.3 (2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	~~≥ 432 volts~~	≥ 432 volts	
7. AUXILIARY FEEDWATER (AFAS)			
a. Manual (Trip Buttons)	Not Applicable	Not Applicable	See ITS 3.3.4
b. Automatic Actuation Logic	Not Applicable	Not Applicable	
c. SG 2A & 2B Level Low	≥ 19.0%	≥ 18.0 %	See ITS 3.3.3
8. AUXILIARY FEEDWATER ISOLATION			
a. Steam Generator ΔP – High	≤ 275 psid	89.2 to 281 psid	See ITS 3.3.3
b. Feedwater Header ΔP – High	≤ 150.0 psid	56.0 to 157.5 psid	

M01

Add proposed ITS 3.3.5 Applicability:
When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
6. LOSS OF POWER (LOV)					
SR 3.3.5.3 a. 4.16 kV and 480 V Emergency Bus Undervoltage (Loss of Voltage)	SFCP	SFCP	SFCP	1, 2, 3, 4	
SR 3.3.5.3 b. 4.16 kV and 480 V Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3, 4	
7. AUXILIARY FEEDWATER (AFAS)					
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	1, 2, 3	See ITS 3.3.4
b. SG Level (A/B) – Low	SFCP	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1), SFCP(2)	1, 2, 3	See ITS 3.3.4
8. AUXILIARY FEEDWATER ISOLATION					
a. SG Level (A/B) – Low and SG Differential Pressure (B to A/A to B) – High	N.A.	SFCP	SFCP	1, 2, 3	See ITS 3.3.3
b. SG Level (A/B) – Low and Feedwater Header Differential Pressure (B to A/A to B) – High	N.A.	SFCP	SFCP	1, 2, 3	

TABLE NOTATION

(1) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay (solid-state component) and verification of the OPERABILITY of each initiation relay (solid-state component).

(2) An actuation relay test shall be performed which shall include the energization/de-energization of each actuation relay and verification of the OPERABILITY of each actuation relay.

(3) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Testing of the ESFAS subgroup relays shall be performed in accordance with the Surveillance Frequency Control Program.

See ITS 3.3.4

DISCUSSION OF CHANGES
ITS 3.3.5, DIESEL GENERATOR (DG) – LOSS OF VOLTAGE START (LOVS)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 *(Category 4 – Relaxation of Required Action)* CTS Table 3.3-3 Actions describe the Actions to be taken when diesel generator (DG) loss of voltage start (LOVS) Function instrument channels are inoperable. ITS 3.3.5 also describes Actions to be taken when DG – LOVS Function instrument channels are inoperable and contains a note that separate condition entry is allowed for each DG – LOVS Function. This changes the CTS by adding a Note stating that separate condition entry is allowed for each Function, those Functions being DG – LOVS 4.16 kV and 480 V loss of voltage and degraded voltage Functions.

The purpose of the CTS Actions is to provide the appropriate compensatory actions for inoperable DG-LOVS Functions. This proposed change will allow separate condition entry for each DG-LOVS Function. The Note clarifies that DG-LOVS Functions are treated as separate entities, each with separate Completion Times. The Conditions of this Specification may be entered independently for each DG-LOVS Function in the LCO. Completion Times for the inoperable channel of a Function will be tracked separately. These changes are acceptable since the proposed Required Actions provide sufficient time to satisfy the Required Actions. Inoperable instrument channels are normally found one at a time, not concurrently. Therefore, the Actions to be taken when DG-LOVS Function instrument channels are inoperable apply as each Function is found to be inoperable and not at the same time. This change is designated as an administrative change and is acceptable because it does not result in technical changes to the CTS.

A03 Unit 1 CTS 4.3.2.1.3 and Unit 2 CS 4.3.2.3 require ENGINEERED SAFETY FEATURES RESPONSE TIME testing of "each" ESFAS function. ITS 3.3.5 does not include response time testing for the DG-LOVS Instrumentation Functions. This changes the CTS by clearly identifying that the ENGINEERED SAFETY FEATURES RESPONSE TIME testing does not apply to the DG-LOVS Instrumentation Functions.

The purpose of Unit 1 CTS 4.3.2.1.3 and Unit 2 CTS 4.3.2.3 is to ensure that the actuation response times, when applicable to a Function, are less than or equal to the maximum values assumed in the accident analysis. The Unit 1 UFSAR Table 13.8.2-2, states that the ENGINEERED SAFETY FEATURES RESPONSE TIME testing is not applicable to the 4.16 kV Loss of Voltage and Degraded Voltage Functions, and the 480 V Degraded Voltage Function. The Unit 2 UFSAR Table 13.7.2-2, states that the ENGINEERED SAFETY FEATURES RESPONSE TIME testing is not applicable to the 4.16 kV Loss of Voltage and

Degraded Voltage Functions, and the 480 V Loss of Voltage and Degraded Voltage Functions. The response time surveillance test acceptance criteria for Technical Specification 3/4.3.2 was relocated by License Amendment No. 128 (Unit 1) and License Amendment No. 67 (Unit 2) and NRC Safety Evaluation dated July 12, 1994. The measurement of engineered safety features actuation systems instrumentation response times at the specified frequencies provides assurance that the ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit is taken in the analyses for those channels with response times indicated as not applicable. Therefore, this change is acceptable since ENGINEERED SAFETY FEATURES RESPONSE TIME testing of the Loss of Voltage and Degraded Voltage Functions is not required. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS Table 3.3-3 for ESFAS instrumentation has three columns stating various requirements for each function. These columns are labeled, "TOTAL NO. OF CHANNELS," "CHANNELS TO TRIP," and "MINIMUM CHANNELS OPERABLE." ITS 3.3.5 does not retain the "TOTAL NO. OF CHANNELS" or "CHANNELS TO TRIP" columns. DOC LA01 describes the change that moves the information of the "TOTAL NO. OF CHANNELS" and "CHANNELS TO TRIP" columns to the Bases. This changes the CTS by changing the "MINIMUM CHANNELS OPERABLE" column to "REQUIRED CHANNELS OPERABLE" and changes the number of channels to 2 per bus for Unit 1 Functional Units 6.a, 6.b, and 6.c, and changes the number of channels to 2 per bus for Unit 2 Functional Unit 6.a.(1) and to 3 per bus for Unit 2 Functional Units 6.a.(2), 6.b.(1), and 6.b.(2). Additionally, the revised "MINIMUM CHANNELS OPERABLE" criteria are moved to the ITS 3.3.5 LCO statement and the "MINIMUM CHANNELS OPERABLE" column is deleted from CTS Table 3.3-3. This change is designated as administrative because it does not result in a technical change to the CTS. | R1

MORE RESTRICTIVE CHANGES

M01 CTS Tables 3.3-3 and 4.3-2 requirements for the Degraded Voltage and Loss of Voltage Functional Units are applicable in MODES 1, 2, and 3. ITS 3.3.5 requires the Degraded Voltage and Loss of Voltage Functions to be OPERABLE in MODES 1, 2, 3, and 4, and when the associated EDG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown." This changes the CTS by expanding the conditions under which the Degraded Voltage and Loss of Voltage Functions must be OPERABLE.

This change is acceptable because requiring the Degraded Voltage and Loss of Voltage Functions to be OPERABLE when LCO 3.8.2 requires an EDG to be OPERABLE ensures that the automatic loss of power start of the EDG is available when needed. This change is designated as more restrictive because it expands the applicability in which equipment is required to be OPERABLE.

M02 **Unit 2 only:** *(Category 3 - Relaxation of Completion Time)* CTS Table 3.3-3 Action 17A requires that, with the number of OPERABLE channels one less than the total number of channels per DG, restore the inoperable channel to

OPERABLE status within 48 hours (2 channels per DG Function) or place the inoperable channel in trip within 1 hour. Action 17A applies to the 2 channels per DG 4.16 kV Loss of Voltage Function. CTS Table 3.3-3 Action 17B requires that, with the number of OPERABLE channels one less than the total number of channels per DG, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time (3 channels per DG Functions), or place the inoperable channel in trip within 1 hour. Action 17B applies to the 3 channels per DG 4.16 kV Degraded Voltage Function, 3 channels per DG 480 Loss of Voltage Function, and 3 channels per DG 480V Degraded Voltage Function.

ITS 3.3.5 retains the current licensing basis with time allotted to restore the inoperable channel to OPERABLE status before requiring the inoperable channel be placed in trip. ITS 3.3.5 ACTION A is consistent with CTS Action 17A. ITS 3.3.5 ACTION B is consistent with CTS Action 17B. The CTS Actions use an "or" connector. ITS ACTIONS that use an "OR" connector use the same Completion Time. This changes the CTS by eliminating the allowance to place the channel in trip within 1 hour after the channel is not restored within the required Completion Time and instead requires the channel to be placed in trip within 48 hours (ACTION A), and within the same 48 hours or in accordance with the Risk Informed Completion Time (ACTION B), consistent with the time currently allotted to restore the inoperable channel to OPERABLE status. If the actions (i.e., restore or trip the inoperable channel) cannot be completed within the required Completion Time, the associated DG must be immediately declared inoperable. This change is acceptable because if the inoperable channel is not restored to OPERABLE status within the associated Completion Time, the channel is placed in trip. However, the ITS 3.3.5 time to trip the channel is 1 hour less than the time allotted in CTS. If the inoperable channel is not restored to OPERABLE status with 48 hours (ACTION A), or within 48 hours or in accordance with the Risk Informed Completion Time (ACTION B), the DG is immediately declared inoperable (ACTION D). This change is designated as more restrictive because it reduces the Completion Time for a CTS Required Action.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS 3.3.2.1 and Unit 2 CTS 3.2.1 Table 3.3-4, ESFAS Instrumentation Trip Values, has two columns stating trip setpoints and allowable values for each Function. Unit 1 CTS 3.3.2.1 and Unit 2 CTS 3.2.1 require the ESFAS instrumentation setpoints to be set consistent with the Trip Setpoint values shown in Table 3.3-4. However, the CTS Action is only required to be taken when the setpoint is less conservative than the Allowable Value column of Table 3.3-4. When the setpoint is less conservative than the Allowable Value, the channel is to be declared inoperable and adjusted consistent with the

Trip Setpoint value. CTS Table 3.3-4 specifies both the Trip Setpoints and Allowable Values for the ESFAS Actuation Instrumentation DG-LOVS Functional Units. ITS 3.3.5 requires the instrumentation for each DG-LOVS Function in the LCO to be OPERABLE. ITS SR 3.3.5.3 specifies only the Allowable Values for the DG-LOVS Instrumentation Functions. The Allowable Values represent the OPERABILITY limit of the channels in ITS. This changes the CTS by moving the Trip Setpoints to the Technical Requirements Manual (TRM).

The purpose of the trip setpoint requirements is to ensure required automatic safety systems are actuated to protect against violating core design limits, breaching the Reactor Coolant System pressure boundary, and to mitigate accidents. Pursuant to 10 CFR 50.36(c)(1)(ii)(A), if it is determined that an automatic protective device for a variable on which a safety limit has been placed (i.e., limiting safety system setting) does not function as required, appropriate action is taken to ensure the abnormal situation is corrected before a safety limit is exceeded, which may include shutting down the reactor. The PSL Instrument Setpoint Methodology calculates nominal trip setpoints (NTSPs) using methods consistent with the guidance provided in NRC Regulatory Guide 1.105, “Setpoints For Safety-Related Instrumentation,” and ANSI/ISA Standard 67.04, "Setpoints for Nuclear Safety-Related Instrumentation." Additionally, pre-defined limits (double-sided Operability limits and as-left limits) are determined for each instrument consistent with the guidance provided in NRC Regulatory Guide 1.105 and ANSI/ISA–RP67.04, “Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation.” The instrument Operability limit band in plant uncertainty calculations is synonymous with the as-found acceptance criteria band specified in ITS and is centered about the nominal equipment setting (clarified in calculations as the field trip setpoint or NTSP). The PSL Instrument Setpoint Methodology, including the method of determining instrument uncertainties, was reviewed by the NRC during the review of Extended Power Uprate (EPU) as documented in the EPU safety evaluations for License Amendments 213 (Unit 1) and 163 (Unit 2), dated July 9, 2012 and September 24, 2012, respectively (NRC ADAMS Accession Nos. ML12181A019 and ML12235A463). The NRC staff determined that the PSL Instrument Setpoint Methodology described in the EPU license request meets the requirements of 10 CFR 50, Appendix A, General Design Criterion 13 and the regulatory guidance in RG 1.105 (EPU SE pg. 92 (Unit 1) and 95 (Unit 2)). The NRC staff also determined that the application of the clarifications of RG 1.105 in Regulatory Issue Summary 2006-17 “NRC Staff Position on the Requirements of 10 CFR 50.36, ‘Technical Specifications,’ Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels,” is acceptable (EPU SE pg. 94 (Unit 1) and 97 (Unit 2)). | R1

The removal of these details for meeting Technical Specification requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the Allowable Values associated with the DG-LOVS Instrumentation, which are designated as the Operability limits for the required instrument Functions. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is

designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 3.3-3 for ESFAS instrumentation has three columns stating various requirements for each function. These columns are labeled, "TOTAL NO. OF CHANNELS," "CHANNELS TO TRIP," and "MINIMUM CHANNELS OPERABLE." ITS 3.3.5 does not retain the "TOTAL NO. OF CHANNELS" or "CHANNELS TO TRIP" columns. This changes the CTS by moving the information of the "TOTAL NO. OF CHANNELS" and "CHANNELS TO TRIP" columns to the Bases. The MINIMUM CHANNELS OPERABLE" column is deleted as described in discussion of change (DOC) A03. | R1

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS 3.3.5 LCO still retains the requirement for the number of required channels and the appropriate Condition to enter if a required channel becomes inoperable. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS Table 3.3-3 Functional Units 6.a - 4.16 kV Loss of Voltage, 6.b – 4.16 kV Degraded Voltage, and 6.c – 480 V Degraded Voltage require 2 channels per DG for each Function. Unit 2 CTS Table 3.3-3 Functional Units 6.a.(1) - 4.16 kV Loss of Voltage requires 2 channels per DG. Unit 2 CTS Functional Units 6.a.(2) 480 V Loss of Voltage, 6.b.(1) – 4.16 kV Degraded Voltage, and 6.b.(2) – 480 V Degraded Voltage each require 3 channels per DG. However, CTS Table 3.3-3 only describes the Actions to be taken when one channel per DG is inoperable for a DG-LOVS Function. ITS LCO 3.3.5 states the DG-LOVS Functions and maintains the same operable channels per DG requirements as CTS. ITS also provides actions to be taken when one diesel generator (DG) loss of voltage start (LOVS) Function instrument channel is inoperable (Unit 1 ACTION A and Unit 2 ACTIONS A and B). Note that the Unit 2 DG-LOVS Function channel operability requirements differ for the Unit 2 CTS Functional Units: 6.a.(1)) - 4.16 kV Loss of Voltage requires 2 channels per DG (Unit 2 ACTION A), and Functional Units 6.a.(2) 480 V Loss of Voltage, 6.b.(1) – 4.16 kV Degraded Voltage, and 6.b.(2) – 480 V Degraded Voltage each require 3 channels per DG (Unit 2 ACTION B).

Additionally, ITS provides Actions for one or more DG-LOVS Functions with two channels per DG inoperable requiring one inoperable channel be restored to OPERABLE status within 1 hour (Unit 1 ACTION B and Unit 2 ACTION C). ITS also provides Actions if the Required Actions and Associated Completion Times are not met requiring the associated DG be declared inoperable immediately (Unit 1 ACTION C and Unit 2 ACTION D). For CTS, if one or more Functions with two or more channels per DG are inoperable, or if the Required Action is not Completed in the associated Completion Time, CTS 3.0.3 is entered since there is no Condition and Action specified in the CTS. This changes the CTS by explicitly providing an ACTION for one or more Functions with two or more channels per DG inoperable, and an ACTION if the Required Action is not Completed in the associated Completion Time.

The purpose of CTS is to ensure that the LOVS-DG logic is in a known configuration either by restoring the inoperable channel to OPERABLE status or by placing the inoperable channel in trip. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the allowed Completion Time. ITS 3.3.5 retains the CTS requirement to either restore the inoperable channel to OPERABLE status or place the channel in trip when one or more Functions with one channel per DG inoperable. Additionally, ITS provides Actions for one or more DG-LOVS Functions with two channels per DG inoperable requiring one inoperable channel be restored to OPERABLE status within 1 hour (Unit 1 ACTION B and Unit 2 ACTION C). ITS also provides Actions if the Required Actions and Associated Completion Times are not met requiring the associated DG be declared inoperable immediately (Unit 1 ACTION C and Unit 2 ACTION D). This change is designated as less restrictive because the ITS Actions are less restrictive than the CTS Actions.

L02 **Unit 1 only:** *(Category 4 - Relaxation of Required Action)* CTS Table 3.3-3 Action 12 states, in part, that with the number of OPERABLE channels one less than the total number of channels, "operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST." This CTS Action applies to the Loss of Voltage and Degraded Voltage Functions of CTS Table 3.3-3. ITS 3.3.5 ACTION A is the applicable action for the Loss of Voltage and Degraded Voltage Functions when one channel is inoperable, and does not include the restoration time limit of "until performance of the next required CHANNEL FUNCTIONAL TEST." This changes the CTS by allowing operation with an inoperable channel for an unlimited amount of time provided the inoperable channel is in the tripped condition.

The purpose of CTS Table 3.3-3 Action 12 is to only allow operation until performance of the next required CHANNEL FUNCTIONAL TEST. This requirement is based upon the assumption that when it is time to test the other OPERABLE channels in the associated Function, the OPERABLE channels cannot be tested with the inoperable channel in trip. However, CTS 3.0.6 (ITS LCO 3.0.5) is a generic allowance that will allow the inoperable channel to be restored to service in order to perform Surveillances on the other OPERABLE

channels in the associated Function. Thus, using this generic allowance, it is possible to test the remaining OPERABLE channels in the associated Function, and there is no reason to restrict the generic allowance from applying to these specific channels. As such, the CTS Table 3.3-3 Action 12 statement is not necessary and has been deleted. The administrative controls required by ITS LCO 3.0.5 will ensure the time the channel is returned to service in conflict with the requirements of ITS 3.3.5 ACTION A is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY of the other channels. In addition, this specific example (taking an inoperable channel out of the tripped condition) is discussed in the Bases of ISTS SR 3.0.5. Therefore, this change is acceptable for the reasons described above. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.3 INSTRUCTIONS ~~(Analog)~~

3.3.~~6~~ Diesel Generator (DG) - Loss of Voltage Start (LOVS) ~~(Analog)~~

LCO 3.3.~~6~~ ~~[Four] channels of Loss of Voltage Function and [four] channels of Degraded Voltage Function auto-initiation instrumentation per DG shall be OPERABLE.~~

INSERT 1

3.3.2.1
Table 3.3-3
Functional Unit
6.a, 6.b, 6.c

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

Table 3.3-3
Functional Unit
6.a, 6.b, 6.c
DOC M01

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each Function.

DOC A02

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel per DG inoperable.	A.1 Place channel in ~~bypass or~~ trip.	1 hour
	~~AND~~	
	~~A.2.1 Restore channel to OPERABLE status.~~	~~[48] hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~
	~~OR~~	

Table 3.3-3
Action 12

1

INSERT 1

Table 3.3-3
Functional Unit
6.a, 6.b, 6.c

The following DG-LOVS Function instrumentation shall be OPERABLE:

a. Two channels of 4.16 kV Loss of Voltage Function per DG;

b. Two channels of 4.16 kV Degraded Voltage Function per DG;

c. Two channels of 480 V Degraded Voltage Function per DG.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~A.2.2 [Place the channel in trip.~~	~~48 hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
B. One or more Functions with two channels per DG inoperable.	B.1 ~~Enter applicable Conditions and Required Actions for the associated DG made inoperable by DG - LOVS instrumentation.~~ ~~OR~~ ~~B.2.1 Place one channel in bypass and the other channel in trip.~~ ~~AND~~ ~~B.2.2~~ Restore one channel to OPERABLE status.	1 hour ~~1 hour~~ ~~[48] hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~
~~C. One or more Functions with more than two channels inoperable~~.	~~C.1 Restore all but two channels to OPERABLE status.~~	~~1 hour~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~

ACTIONS (continue)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D~~. Required Action and associated Completion Time not met.	~~D~~.1 ~~Enter applicable Conditions and Required Actions for the~~ associated DG ~~made~~ inoperable ~~by DG - LOVS instrumentation~~.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.~~6~~.1	~~[~~ Perform CHANNEL CHECK.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~]
SR 3.3.~~6~~.2	Perform CHANNEL FUNCTIONAL TEST.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.~~6~~.3 5	Perform CHANNEL CALIBRATION with setpoint Allowable Values as follows: a. ~~Degraded Voltage Function ≥ [3180] V and ≤ [3220] V~~ ~~Time delay: ≥ [ ] seconds and ≤ [ ] seconds at [ ] V and~~ b. ~~Loss of Voltage Function ≥ [3180] V and ≤ [3220] V~~ ~~Time delay: ≥ [ ] seconds and ≤ [ ] seconds at [ ] V.~~	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.3.2.1.1 Table 4.3-2 Function 6.a, 6.b, 6.c

Table 3.3-4 Function 6.a

a. 4.16 kV Loss of Voltage Function ≥ 2900 V

Time delay: 1 ± 0.5 seconds

Table 3.3-4 Function 6.b

b. 4.16 kV Degraded Voltage Function ≥ 3831 V

Time delay: 18 ± 2 seconds

Table 3.3-4 Function 6.c

c. 480 V Degraded Voltage Function ≥ 415 V

Time Delay: ≤ 9 seconds

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~6~~ Diesel Generator (DG) - Loss of Voltage Start (LOVS) ~~(Analog)~~

3.3.2.1 Table 3.3-3 Functional Unit 6.a.(1), 6.a.(2), 6.b.(1), 6.b.(2)

LCO 3.3.~~6~~ ~~[Four] channels of Loss of Voltage Function and [four] channels of Degraded Voltage Function auto-initiation instrumentation per DG shall be OPERABLE.~~

Table 3.3-3 Functional Unit 6.a.(1), 6.a.(2), 6.b.(1), 6.b.(2) DOC M01

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

----------NOTE----------
DOC A02 Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-3 Action 17A A. One ~~or more~~ Function~~s~~ with one channel per DG inoperable.	A.1 Place channel in ~~bypass or~~ trip. ~~AND~~ A.2~~.1~~ Restore channel to OPERABLE status. ----------NOTE---------- One channel may be bypassed for up to 2 hours for surveillance testing. --- ~~OR~~	~~1 hour~~ 48 hours [48~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~

INSERT 1

INSERT 2

1

INSERT 1

Table 3.3-3
Functional Unit
6.a.(1), 6.a.(2),
6.b.(1), 6.b.(2)

The following DG-LOVS Function instrumentation shall be OPERABLE:

a. Two channels of 4.16 kV Loss of Voltage Function per DG;

b. Three channels of 4.16 kV Degraded Voltage Function per DG;

c. Three channels of 480 V Loss of Voltage Function per DG;

d. Three channels of 480 V Degraded Voltage Function per DG.

1

INSERT 2

Table 3.3-3 Action 17B	B. One or more Functions with one channel per DG inoperable (except 4.16 kV Loss of Voltage Function).	B.1	----------NOTE---------- One channel may be bypassed for up to 2 hours for surveillance testing. ---------- Place channel in trip.	48 hours OR In accordance with the Risk Informed Completion Time Program
		OR		
		B.2	Restore channel to OPERABLE status.	48 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~A.2.2 [Place the channel in trip.~~	~~48 hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
~~B~~. One or more Functions with two channels per DG inoperable.	~~B~~.1 ~~Enter applicable Conditions and Required Actions for the associated DG made inoperable by DG - LOVS instrumentation~~. ~~OR~~ ~~B.2.1 Place one channel in bypass and the other channel in trip.~~ ~~AND~~ ~~B.2.2~~ Restore one channel to OPERABLE status.	1 hour ~~1 hour~~ ~~[48] hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~
~~C. One or more Functions with more than two channels inoperable~~.	~~C.1 Restore all but two channels to OPERABLE status.~~	~~1 hour~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~

ACTIONS (continue)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D~~. Required Action and associated Completion Time not met.	~~D~~.1 ~~Enter applicable Conditions and Required Actions for the~~ associated DG ~~made~~ inoperable ~~by DG - LOVS instrumentation~~.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.~~6~~.1	~~[~~ Perform CHANNEL CHECK.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~]
SR 3.3.~~6~~.2	Perform CHANNEL FUNCTIONAL TEST.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.~~6~~.3 5	Perform CHANNEL CALIBRATION with setpoint Allowable Values as follows: ~~a. Degraded Voltage Function ≥ [3180] V and ≤ [3220] V~~ ~~Time delay: ≥ [ ] seconds and ≤ [ ] seconds at [ ] V and~~ ~~b. Loss of Voltage Function ≥ [3180] V and ≤ [3220] V~~ ~~Time delay: ≥ [ ] seconds and ≤ [ ] seconds at [ ] V.~~	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.3.2.1.1 Table 4.3-2 Function 6.a, 6.b

2 1 1 2 1 1

Table 3.3-4 Function 6.a.(1)
Table 3.3-4 Function 6.a.(2)
Table 3.3-4 Function 6.b.(1)
Table 3.3-4 Function 6.b.(2)

a. 4.16 kV Loss of Voltage Function ≥ 3120 V
b. 480 V Loss of Voltage Function ≥ 360 V
c. 4.16 kV Degraded Voltage Function ≥ 3848 V
Time Delay: < 10 seconds
d. 480 V Degraded Voltage Function ≥ 432 V

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.5, DIESEL GENERATOR (DG) – LOSS OF VOLTAGE START (LOVS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The heading for ISTS 3.3.6 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2). Therefore, ISTS 3.3.2 is not included in the PSL ITS. The successive Specifications are renumbered as necessary.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~6~~ Diesel Generator (DG) - Loss of Voltage Start (LOVS) ~~(Analog)~~

5

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or insufficiently stable to allow safe plant operation. Undervoltage protection will generate a LOVS in the event a Loss of Voltage or Degraded Voltage condition occurs. There are two LOVS Functions for each 4.16 kV vital bus.

INSERT 1

~~Four undervoltage relays with inverse time characteristics are provided on each 4.16 kV Class 1E instrument bus for the purpose of detecting a sustained undervoltage condition or a loss of bus voltage. The relays are combined in a two-out-of-four logic to generate a LOVS if the voltage is below 75% for a short time or below 90% for a long time. The LOVS initiated actions are described in Reference 1.~~

1

Trip Setpoints and Allowable Values

The trip setpoints and Allowable Values are based on the analytical limits presented in Reference 2. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Values specified in SR 3.3.~~6~~.3 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 3. The actual nominal trip setpoint is normally still more conservative than that required by the plant specific setpoint calculations. If the measured setpoint does not exceed the documented surveillance trip acceptance criteria, the undervoltage relay is considered OPERABLE.

5

5

Setpoints in accordance with the Allowable Values will ensure that the consequences of accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the accident and the equipment functions as designed.

~~The undervoltage protection scheme has been designed to protect the plant from spurious trips caused by the offsite power source. This is made possible by the inverse voltage time characteristics of the relays used.~~ A complete loss of offsite power will result in approximately a 1 second delay in LOVS actuation. The DG starts and is available to

1

1

INSERT 1

The 4.16 kV LOVS Functions include Loss of Voltage and Degraded Voltage Functions. The allowable values are provided in SR 3.3.5.3.

Each Class 1E 4.16 kV Bus utilizes two undervoltage definite time delay relays, in a two-out-of-two coincident logic for loss of voltage detection. The function of these relays is to initiate source disconnection, load shedding, diesel generator starting, and load sequencing on the effected train (bus). The LOVS initiated actions are described in Reference 1. R1

Each Class 1E 4.16 kV Bus utilizes two sets of 2 undervoltage definite time delay relays in a two-out-of-two coincident logic for degraded voltage detection. The function of these relays is the same as described for the loss of voltage detection. R1

Each Class 1E 480V Bus utilizes two undervoltage definite time delay relays in a two-out-of-two coincident logic scheme for degraded voltage detection. R1

BASES

BACKGROUND (continued)

accept loads within a 10 second time interval on the Engineered Safety Features Actuation System (ESFAS) or LOVS. Emergency power is established within the maximum time delay assumed for each event analyzed in the accident analysis (Ref. 2).

INSERT 2 →

~~Since there are four protective channels in a two-out-of-four trip logic for each division of the 4.16 kV power supply, no single failure will cause or prevent protective system actuation. This arrangement meets IEEE Standard 279-1971 criteria (Ref. 4).~~

1

APPLICABLE SAFETY ANALYSES

The DG - LOVS is required for Engineered Safety Features (ESF) systems to function in any accident with a loss of offsite power. Its design basis is that of the ESFAS.

Accident analyses credit the loading of the DG based on a loss of offsite power during a loss of coolant accident. The actual DG start has historically been associated with the ESFAS actuation. The diesel loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. ~~The analysis assumes a nonmechanistic DG loading, which does not explicitly account for each individual component of the loss of power detection and subsequent actions.~~ This delay time includes contributions from the DG start, DG loading, and Safety Injection System component actuation. The response of the DG to a loss of power must be demonstrated to fall within this analysis response time when including the contributions of all portions of the delay.

1

The required channels of LOVS, in conjunction with the ESF systems powered from the DGs, provide plant protection in the event of any of the analyzed accidents discussed in Reference 2, in which a loss of offsite power is assumed. LOVS channels are required to meet the redundancy and testability requirements of GDC 21 in 10 CFR 50, Appendix A (Ref. ~~5~~).

6

The delay times assumed in the safety analysis for the ESF equipment include the ~~[~~10~~]~~ second DG start delay and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment include the appropriate DG loading and sequencing delay.

The DG - LOVS channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

two DG

The LCO for the LOVS requires that ~~four~~ channels per ~~bus~~ of each LOVS instrumentation Function be OPERABLE in MODES 1, 2, 3, and 4 and when the associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown." The LOVS supports safety systems associated

1

1

INSERT 2

The Class 1E 4.16 kV system and Class 1E 480 V system are designed to meet the single failure criteria as defined in IEEE 308 (Ref. 4) and IEEE 279 (Ref. 5). Specifically, any credible single failure shall not prevent the satisfactory performance of the minimum Class 1E 4.16 kV loads or Class 1E 480 V loads required to achieve and maintain safe shutdown or to monitor or mitigate the consequences of any design basis accident or event. The worst case single failure will disable only one train of the Class 1E 4.16 kV or Class 1E 480 V system. | R1

BASES

LCO (continued)

with the ESFAS. In MODES 5 and 6, the ~~four~~ two channels must be OPERABLE whenever the associated DG is required to be OPERABLE to ensure that the automatic start ovf the DG is available when needed.

~~Actions allow maintenance (trip channel) bypass of individual channels. Plants are restricted to 48 hours in a trip channel bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic). At plants where adequate channel to channel independence has been demonstrated, specific exceptions have been approved by the NRC staff to permit one of the two-out-of-four channels to be bypassed for an extended period of time.~~ 1

~~Loss of LOVS Function could result in the delay of safety system initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power, which is an anticipated operational occurrence, the DG powers the motor driven auxiliary feedwater pumps. Failure of these pumps to start would leave only the one turbine driven pump as well as an increased potential for a loss of decay heat removal through the secondary system.~~ 1

Only Allowable Values are specified for each Function in the LCO. Nominal trip setpoints are specified in the plant specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within the Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant specific setpoint calculation. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

~~[For this unit, the Bases for the Allowable Values and trip setpoints are as follows:]~~ 1

APPLICABILITY

The DG - LOVS actuation Function is required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE, so that it can perform its function on a loss of power or degraded power to the vital bus.

ACTIONS

A LOVS channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint

BASES

ACTIONS (continued)

analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. Determination of setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the instrument is set up for adjustment to bring it within specification. If the actual trip setpoint is not within the Allowable Value, the channel is inoperable and the appropriate Conditions must be entered.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the channel is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition entered. The required channels are specified on a per DG basis.

~~When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be entered immediately if applicable in the current MODE of operation.~~

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this LCO may be entered independently for each Function. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

A.1~~, A.2.1, and A.2.2~~

Condition A applies if one channel is inoperable for one or more Functions per DG bus.

If the channel cannot be restored to OPERABLE status, the affected channel should ~~either~~ be ~~bypassed or~~ tripped within 1 hour (Required Action A.1).

the this

Placing ~~this~~ channel in ~~either~~ Condition ensures that logic is in a known configuration. In trip, the LOVS Logic is one-out-of-~~three~~. ~~In bypass, the LOVS Logic is two-out-of-three.~~ The 1 hour Completion Time is sufficient to perform the~~se~~ Required Action~~s~~.

one

~~Once Required Action A.1 has been complied with, Required Action A.2.1 allows [48] hours to repair the inoperable channel for those plants that have not demonstrated sufficient channel to channel independence on this Function. [Alternatively, a Completion Time can be determined in~~

BASES

ACTIONS (continued)

~~accordance with the Risk Informed Completion Time Program.] If the channel cannot be restored to OPERABLE status, it must be tripped in accordance with Required Action A.2.2. The time allowed to repair or trip the channel is reasonable to repair the affected channel while ensuring that the risk involved in operating with the inoperable channel is acceptable. The [48] hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel is a rare event during any given [48] hour period. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 1

B.1~~, B.2.1, and B.2.2~~ 3

Condition B applies if two channels are inoperable for one or more Functions per DG.

either

Condition C is entered and the affected DG is declared inoperable.

If ~~the~~ channel cannot be restored to OPERABLE status within 1 hour, ~~the Conditions and Required Actions for the associated DG made inoperable by DG - LOVS instrumentation are required to be entered. Alternatively, one affected channel is required to be bypassed and the other is tripped, in accordance with Required Action B.2.1. This places the Function in one-out-of-two logic.~~ The 1 hour Completion Time is sufficient to perform the Required Actions. 1 1

~~Once Required Action B.2.1 has been complied with, Required Action B.2.2 allows [48] hours to repair the bypassed or inoperable channel. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 1

After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. ~~Therefore, the channel that is still inoperable after completion of Required Action B.2.2 shall be placed in trip if more than [48] hours have elapsed since the initial channel failure. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 1

~~C.1~~

~~Condition C applies when more than two undervoltage or Degraded Voltage channels on a single bus are inoperable.~~ 1

BASES

ACTIONS (continued)

~~Required Action C.1 requires all but two channels to be restored to OPERABLE status within 1 hour [or in accordance with the Risk Informed Completion Time Program]. With more than two channels inoperable, the logic is not capable of providing a DG - LOVS signal for valid Loss of Voltage or Degraded Voltage conditions. The 1 hour Completion Time is reasonable to evaluate and take action to correct the degraded condition in an orderly manner and takes into account the low probability of an event requiring LOVS occurring during this interval.~~

C.1

Condition ~~D~~ applies if the Required Actions and associated Completion Times are not met.

Required Action ~~D~~.1 ~~ensures that Required Actions for~~ the affected DG inoperabi~~liti~~e~~s are initiated~~. Depending upon plant MODE, the actions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2 are required immediately.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each DG - LOVS Function.

~~[~~ SR 3.3.~~6~~.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the indicated output of the potential transformers that feed the LOVS undervoltage relays. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two channels could be an indication of excessive drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If the channels are within the criteria, it is an indication that the channels are OPERABLE. ~~]~~

~~[The Frequency, about once every shift, is based upon operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The~~

SURVEILLANCE REQUIREMENTS (continued)

~~CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 2

5
SR 3.3.~~6~~.2 3

A CHANNEL FUNCTIONAL TEST is performed to ensure that the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

~~[The Frequency of [92] days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any [92] day Frequency is a rare event. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. ~~There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.~~

SR 3.3.~~6~~.3

SR 3.3.~~6~~.3 is the performance of a CHANNEL CALIBRATION. The CHANNEL CALIBRATION verifies the accuracy of each component within the instrument channel. This includes calibration of the undervoltage relays and demonstrates that the equipment falls within the specified operating characteristics defined by the manufacturer.

The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. The~~re is a plant specific program which~~ verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The setpoints, as well as the response to a Loss of Voltage and Degraded Voltage test, shall include a single point verification that the trip occurs within the required delay time as shown in Reference 1. ~~[The Frequency is based upon the assumption of an [18] month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

BASES

REFERENCES

1. FSAR, Section ~~[~~8.3~~]~~.
2. FSAR, Chapter ~~[~~15~~]~~.
3. ~~"Plant Protection System Selection of Trip Setpoint Values."~~
4. IEEE Standard 279-1971.
5. 10 CFR 50, Appendix A, GDC 21.

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~6~~ Diesel Generator (DG) - Loss of Voltage Start (LOVS) ~~(Analog)~~

5

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or insufficiently stable to allow safe plant operation. Undervoltage protection will generate a LOVS in the event a Loss of Voltage or Degraded Voltage condition occurs. There are two LOVS Functions for each 4.16 kV vital bus.

INSERT 1

~~Four undervoltage relays with inverse time characteristics are provided on each 4.16 kV Class 1E instrument bus for the purpose of detecting a sustained undervoltage condition or a loss of bus voltage. The relays are combined in a two-out-of-four logic to generate a LOVS if the voltage is below 75% for a short time or below 90% for a long time. The LOVS initiated actions are described in Reference 1.~~

1

Trip Setpoints and Allowable Values

The trip setpoints and Allowable Values are based on the analytical limits presented in Reference 2. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Values specified in SR 3.3.~~6~~.3 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 3. The actual nominal trip setpoint is normally still more conservative than that required by the plant specific setpoint calculations. If the measured setpoint does not exceed the documented surveillance trip acceptance criteria, the undervoltage relay is considered OPERABLE.

5

5

Setpoints in accordance with the Allowable Values will ensure that the consequences of accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the accident and the equipment functions as designed.

~~The undervoltage protection scheme has been designed to protect the plant from spurious trips caused by the offsite power source. This is made possible by the inverse voltage time characteristics of the relays used.~~ A complete loss of offsite power will result in approximately a 1 second delay in LOVS actuation. The DG starts and is available to

1

1

INSERT 1

The 4.16 kV LOVS Functions include Loss of Voltage and Degraded Voltage Functions. The allowable values are provided in SR 3.3.5.3.

Each Class 1E 4.16 kV Bus utilizes two solid-state undervoltage relays, in a two-out-of-two coincident logic, for loss of voltage detection. The function of these relays is to initiate source disconnection, load shedding, diesel generator starting, and load sequencing on the effected train (bus). The LOVS initiated actions are described in Reference 1. | R1

Each Class 1E 4.16 kV Bus utilizes three solid state undervoltage relays in a two-out-of-three coincident logic for degraded voltage detection. The function of these relays is the same as described for the loss of voltage detection. | R1

Each Class 1E 480V Bus utilizes three solid state undervoltage relays in a two-out-of-three coincident logic scheme, for loss of voltage detection. | R1

Each Class 1E 480V Bus utilizes three solid state undervoltage relays in a two-out-of-three coincident logic scheme, for degraded voltage detection. | R1

BASES

BACKGROUND (continued)

accept loads within a 10 second time interval on the Engineered Safety Features Actuation System (ESFAS) or LOVS. Emergency power is established within the maximum time delay assumed for each event analyzed in the accident analysis (Ref. 2).

INSERT 2 →

~~Since there are four protective channels in a two-out-of-four trip logic for each division of the 4.16 kV power supply, no single failure will cause or prevent protective system actuation. This arrangement meets IEEE Standard 279-1971 criteria (Ref. 4).~~

1

APPLICABLE SAFETY ANALYSES

The DG - LOVS is required for Engineered Safety Features (ESF) systems to function in any accident with a loss of offsite power. Its design basis is that of the ESFAS.

Accident analyses credit the loading of the DG based on a loss of offsite power during a loss of coolant accident. The actual DG start has historically been associated with the ESFAS actuation. The diesel loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. ~~The analysis assumes a nonmechanistic DG loading, which does not explicitly account for each individual component of the loss of power detection and subsequent actions.~~ This delay time includes contributions from the DG start, DG loading, and Safety Injection System component actuation. The response of the DG to a loss of power must be demonstrated to fall within this analysis response time when including the contributions of all portions of the delay.

1

The required channels of LOVS, in conjunction with the ESF systems powered from the DGs, provide plant protection in the event of any of the analyzed accidents discussed in Reference 2, in which a loss of offsite power is assumed. LOVS channels are required to meet the redundancy and testability requirements of GDC 21 in 10 CFR 50, Appendix A (Ref. ~~5~~).

6

The delay times assumed in the safety analysis for the ESF equipment include the ~~[~~10~~]~~ second DG start delay and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment include the appropriate DG loading and sequencing delay.

The DG - LOVS channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

two channels per DG of the 4.16 kV Loss of Voltage Function; and three channels per DG of the 4.16 kV Degraded Voltage, 480 V Loss of Voltage, and 480 V Degraded Voltage Functions →

The LCO for the LOVS requires that ~~four channels per bus of each LOVS instrumentation Function~~ be OPERABLE in MODES 1, 2, 3, and 4 and when the associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown." The LOVS supports safety systems associated

1

1

INSERT 2

The Class 1E 4.16 kV system and Class 1E 480 V system are designed to meet the single failure criteria as defined in IEEE 308 (Ref. 4) and IEEE 279 (Ref. 5). Specifically, any credible single failure shall not prevent the satisfactory performance of the minimum Class 1E 4.16kV loads or Class 1E 480 V loads required to achieve and maintain safe shutdown or to monitor or mitigate the consequences of any design basis accident or event. The worst case single failure will disable only one train of the Class 1E 4.16 kV or Class 1E 480 V system. | R1

3

5

BASES

LCO (continued)

two channels per DG of the 4.16 kV Loss of Voltage Function; and three channels per DG of the 4.16 kV Degraded Voltage, 480 V Loss of Voltage, and 480 V Degraded Voltage Functions

with the ESFAS. In MODES 5 and 6, the ~~four channels~~ must be OPERABLE whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed.

~~Actions allow maintenance (trip channel) bypass of individual channels. Plants are restricted to 48 hours in a trip channel bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic). At plants where adequate channel to channel independence has been demonstrated, specific exceptions have been approved by the NRC staff to permit one of the two-out-of-four channels to be bypassed for an extended period of time.~~

1

~~Loss of LOVS Function could result in the delay of safety system initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power, which is an anticipated operational occurrence, the DG powers the motor driven auxiliary feedwater pumps. Failure of these pumps to start would leave only the one turbine driven pump as well as an increased potential for a loss of decay heat removal through the secondary system.~~

1

Only Allowable Values are specified for each Function in the LCO. Nominal trip setpoints are specified in the plant specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within the Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant specific setpoint calculation. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

~~[For this unit, the Bases for the Allowable Values and trip setpoints are as follows:]~~

1

APPLICABILITY

The DG - LOVS actuation Function is required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE, so that it can perform its function on a loss of power or degraded power to the vital bus.

ACTIONS

A LOVS channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint

3

1

ACTIONS (continued)

analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. Determination of setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the instrument is set up for adjustment to bring it within specification. If the actual trip setpoint is not within the Allowable Value, the channel is inoperable and the appropriate Conditions must be entered.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the channel is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition entered. The required channels are specified on a per DG basis.

~~When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be entered immediately if applicable in the current MODE of operation.~~ 1

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this LCO may be entered independently for each Function. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

A.1~~, A.2.1, and A.2.2~~ 4.16 kV Loss of Voltage Function channel per DG is inoperable. 1

Condition A applies if one ~~channel is inoperable for one or more Functions per DG bus~~.

within 48 hours

If the channel cannot be restored to OPERABLE status, the affected channel should ~~either~~ be ~~bypassed or~~ tripped within ~~1~~ hour (Required Action A.1). 1

48 s

the this

Placing ~~this~~ channel in ~~either~~ Condition ensures that logic is in a known configuration. In trip, the LOVS Logic is one-out-of-~~three~~. ~~In bypass, the LOVS Logic is two-out-of-three.~~ The ~~1~~ hour Completion Time is sufficient to perform the~~se~~ Required Action~~s~~. 1

48 one

INSERT 3

~~Once Required Action A.1 has been complied with, Required Action A.2.1 allows [48] hours to repair the inoperable channel for those plants that have not demonstrated sufficient channel to channel independence on this Function. [Alternatively, a Completion Time can be determined in~~ 1

1

INSERT 3

The Required Actions are modified by a Note that allows one channel to be bypassed for up to 2 hours for surveillance testing. This allowance is based on the average time required to perform the channel surveillance.

BASES

ACTIONS (continued)

~~accordance with the Risk Informed Completion Time Program.] If the channel cannot be restored to OPERABLE status, it must be tripped in accordance with Required Action A.2.2.~~ The time allowed to repair or trip the channel is reasonable to repair the affected channel while ensuring that the risk involved in operating with the inoperable channel is acceptable. The ~~[~~48~~]~~ hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel is a rare event during any given ~~[~~48~~]~~ hour period. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 1

INSERT 4

C → ~~B~~.1~~, B.2.1, and B.2.2~~ 3

C

Condition ~~B~~ applies if two channels or more are inoperable for one or more Functions per DG.

Condition D is entered and the affected DG is declared inoperable.

If the channel cannot be restored to OPERABLE status within 1 hour, ~~the Conditions and Required Actions for the associated DG made inoperable by DG - LOVS instrumentation are required to be entered. Alternatively, one affected channel is required to be bypassed and the other is tripped, in accordance with Required Action B.2.1. This places the Function in one-out-of-two logic.~~ The 1 hour Completion Time is sufficient to perform the Required Actions. 1 1

~~Once Required Action B.2.1 has been complied with, Required Action B.2.2 allows [48] hours to repair the bypassed or inoperable channel. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 1

or Condition B

After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. ~~Therefore, the channel that is still inoperable after completion of Required Action B.2.2 shall be placed in trip if more than [48] hours have elapsed since the initial channel failure. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 1

~~C.1~~

~~Condition C applies when more than two undervoltage or Degraded Voltage channels on a single bus are inoperable.~~ 1

1

INSERT 4

B.1 and B.2

Table 3.3-3
Action 17B

Condition B applies if one or more Functions with one channel per DG inoperable (except 4.16 kV Loss of Voltage Function). The applicable Functions are the 4.16 kV Degraded Voltage Function, 480 V Loss of Voltage Function, and 480 V Degraded Voltage Function.

If the channel cannot be restored to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time (Required Action B.1), the affected channel is placed in trip (Required Action B.2). Placing the channel in this Condition ensures that logic is in a known configuration. In trip, the LOVS Logic is one-out-of-two. The 48 hour Completion Time is sufficient to perform these Required Actions.

The time allowed to repair or trip the channel is reasonable to repair the affected channel while ensuring that the risk involved in operating with the inoperable channel is acceptable. The 48 hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel is a rare event during any given 48 hour period. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

The Required Actions are modified by a Note that allows one channel to be bypassed for up to 2 hours for surveillance testing. This allowance is based on the average time required to perform the channel surveillance.

BASES

ACTIONS (continued)

~~Required Action C.1 requires all but two channels to be restored to OPERABLE status within 1 hour [or in accordance with the Risk Informed Completion Time Program]. With more than two channels inoperable, the logic is not capable of providing a DG - LOVS signal for valid Loss of Voltage or Degraded Voltage conditions. The 1 hour Completion Time is reasonable to evaluate and take action to correct the degraded condition in an orderly manner and takes into account the low probability of an event requiring LOVS occurring during this interval.~~ 1

D.1 3

Condition D applies if the Required Actions and associated Completion Times are not met. 3

requires declaring

Required Action D.1 ~~ensures that Required Actions for~~ the affected DG inoperabi~~li~~ti~~e~~s~~ are initiated~~. Depending upon plant MODE, the actions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2 are required immediately. 3 1

, "AC Sources - Shutdown", 1

SURVEILLANCE REQUIREMENTS

The following SRs apply to each DG - LOVS Function.

5

~~[~~ SR 3.3.~~6~~.1 2 3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the indicated output of the potential transformers that feed the LOVS undervoltage relays. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two channels could be an indication of excessive drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If the channels are within the criteria, it is an indication that the channels are OPERABLE. ~~]~~ 2

~~[The Frequency, about once every shift, is based upon operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The~~ 2

SURVEILLANCE REQUIREMENTS (continued)

~~CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

5
SR 3.3.~~6~~.2 3

A CHANNEL FUNCTIONAL TEST is performed to ensure that the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

~~[The Frequency of [92] days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any [92] day Frequency is a rare event. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. ~~There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.~~

SR 3.3.~~6~~.3

SR 3.3.~~6~~.3 is the performance of a CHANNEL CALIBRATION. The CHANNEL CALIBRATION verifies the accuracy of each component within the instrument channel. This includes calibration of the undervoltage relays and demonstrates that the equipment falls within the specified operating characteristics defined by the manufacturer.

The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. The~~re is a plant specific program which~~ verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The setpoints, as well as the response to a Loss of Voltage and Degraded Voltage test, shall include a single point verification that the trip occurs within the required delay time as shown in Reference 1. ~~[The Frequency is based upon the assumption of an [18] month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

BASES

REFERENCES

1. FSAR, Section ~~[~~8.3~~]~~.
2. FSAR, Chapter ~~[~~15~~]~~.
3. ~~"Plant Protection System Selection of Trip Setpoint Values."~~ IC-3.17, FPL Setpoint Standard.
4. IEEE Standard 308-1971.
5. ~~4~~. IEEE Standard 279-1971.
6. ~~5~~. 10 CFR 50, Appendix A, GDC 21.

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.5 BASES, DIESEL GENERATOR (DG) – LOSS OF VOLTAGE START (LOVS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The heading for ISTS 3.3.6 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2). Therefore, ISTS 3.3.2 is not included in the PSL ITS. The successive Specifications are renumbered as necessary.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.5, DIESEL GENERATOR (DG) – LOSS OF VOLTAGE START (LOVS)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

3.3.6, Containment Isolation Instrumentation – Refueling

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

INSTRUMENTATION

CONTAINMENT ISOLATION

3/4.3.3 ~~MONITORING~~ INSTRUMENTATION

– REFUELING

A02

~~RADIATION MONITORING~~

LIMITING CONDITION FOR OPERATION

and one automatic actuation logic train

Three containment

LCO 3.3.6 3.3.3.1 ~~The~~ radiation monitor~~ing instrumentation~~ channels ~~shown in Table 3.3-6~~ shall be OPERABLE ~~with their alarm/trip setpoints within the specified limits~~.

Applicability **APPLICABILITY**: ~~As shown in Table 3.3-6.~~

During movement of recently irradiated fuel assemblies within containment.

ACTION:

~~a.~~ ~~With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.~~

M01

A02

required

Condition A b With one or more radiation monitor~~ing~~ channels inoperable, ~~take the ACTION shown in Table 3.3-6.~~

one required automatic actuation logic train inoperable or

~~c. The provisions of Specification 3.0.3 are not applicable.~~

A03

SURVEILLANCE REQUIREMENTS

required containment

SR 3.3.6.1
SR 3.3.6.2
SR 3.3.6.4

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program.

Add proposed SR 3.3.6.3

A02

4.3.3.2 In accordance with the Surveillance Frequency Control Program, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

See ITS 3.3.7

~~TABLE 3.3-6~~

~~RADIATION MONITORING INSTRUMENTATION~~

Allowable Value

~~INSTRUMENT~~	~~MINIMUM CHANNELS OPERABLE~~	~~APPLICABLE MODES~~	~~ALARM/TRIP SETPOINT~~	~~MEASUREMENT RANGE~~	~~ACTION~~	
~~1. AREA MONITORS~~						
a. Fuel Storage Pool Area	1	*	≤ 15 mR/hr	10^{-1} – 10^{4} mR/hr	13	See ITS 3.3.8
LCO 3.3.6 SR 3.3.6.2 — b. Containment ~~(CIS)~~ ← Radiation Monitor	3	****	≤ 90 mR/hr	~~1 – 10^{5} mR/hr~~	~~16~~ ← A	LA01
c. Containment Area – Hi Range	1	1, 2, 3, & 4	≤ 10 R/hr	1 – 10^{7} R/hr	15	See ITS 3.3.9
d. Control Room Isolation — Add Automatic Actuation Logic Function	1 per intake	ALL MODES	≤ 320 cpm	10 - 10^{7} cpm	17	See ITS 3.3.7
						M02
2. PROCESS MONITORS						
a. Containment						
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 – 10^{6} cpm	14	See ITS 3.4.15
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 – 10^{6} cpm	14	

Applicability

* With fuel in the storage pool or building. — See ITS 3.3.8

**** During movement of recently irradiated fuel assemblies within containment.

~~TABLE 3.3-6 (Continued)~~

~~TABLE NOTATION~~

ACTION 12 - DELETED

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s),and

2) Prepare and submit a Special Report to the Commission pursuant to Specification6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

See ITS 3.3.9

ACTION A

~~ACTION 16 -~~ With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, ~~comply with the ACTION requirements of Specification 3.9.9~~.

Add proposed ACTION A

A04

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

INSTRUMENTATION

CONTAINMENT ISOLATION

3/4.3.3 ~~MONITORING~~ INSTRUMENTATION – REFUELING

~~RADIATION MONITORING INSTRUMENTATION~~ A02

LIMITING CONDITION FOR OPERATION

and one automatic actuation logic train

Three containment

LCO 3.3.6 3.3.3.1 ~~The~~ radiation monitor~~ing instrumentation~~ channels ~~shown in Table 3.3-6~~ shall be OPERABLE ~~with their alarm/trip setpoints within the specified limits~~.

Applicability **APPLICABILITY**: ~~As shown in Table 3.3-6.~~ During movement of recently irradiated fuel assemblies within containment.

ACTION:

~~a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.~~ M01

required

Condition A b. With one or more radiation monitor~~ing~~ channels inoperable, take the ~~ACTION shown in Table 3.3-6~~. A02

one required automatic actuation logic train inoperable or

~~c. The provisions of Specification 3.0.3 are not applicable.~~ A03

SURVEILLANCE REQUIREMENTS

required containment

SR 3.3.6.1
SR 3.3.6.2
SR 3.3.6.4

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK~~,~~ CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program. Add proposed SR 3.3.6.3 A02

4.3.3.2 In accordance with the Surveillance Frequency Control Program, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

See ITS 3.3.7

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

Allowable Value

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
~~1. AREA MONITORS~~					
a. Fuel Storage Pool Area					
i. Criticality and Ventilation System Isolation Monitor	4	*	≤ 20 mR/hr	10^{-1} – 10^{4} mR/hr	22
b. Containment ~~Isolation~~	3	****	≤ 90 mR/hr	~~1 – 10^{7} mR/hr~~	~~25~~ A
c. Containment Area – Hi Range	1	1, 2, 3 & 4	Not Applicable	1 - 10^{7} R/hr	27
d. Control Room Isolation	1 per intake	ALL MODES	≤ 320 cpm	10^{-7}– 10^{-2} µCi/cc	26
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10^{-7} – 10^{-2} µCi/cc	23
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 – 10^{7} cpm	23

* With fuel in the storage pool or building.

**** During movement of recently irradiated fuel assemblies within containment.

Radiation Monitor

Add Automatic Actuation Logic Function

LA01

R1

M02

See ITS 3.3.8

See ITS 3.3.9

See ITS 3.3.7

See ITS 3.4.15

See ITS 3.3.8

LCO 3.3.6
SR 3.3.6.2

Applicability

~~TABLE 3.3-6 (Continued)~~

~~ACTION STATEMENTS~~

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

~~ACTION 24 - DELETED~~

ACTION A

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, ~~comply with the ACTION requirements of Specification 3.9.9~~.

Add proposed ACTION A

A04

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

See ITS 3.3.9

DISCUSSION OF CHANGES
ITS 3.3.6, CONTAINMENT ISOLATION INSTRUMENTATION – REFUELING

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.3.3.1 specifies requirements for radiation monitoring instrumentation and list the requirements for each functional unit in CTS Table 3.3-6, including the containment radiation monitoring instrumentation. ITS 3.3.6 provides requirements for the containment isolation instrumentation - refueling instrumentation and provides the requirements in the LCO, Applicability, ACTIONS, and Surveillance Requirements. This changes the CTS by presenting the instrument requirements without use of an instrument table. This change is acceptable as it results solely from the change in the format and presentation of the CTS necessary to conform to the ISTS. As the proposed change is the result of changes in the format and presentation of the CTS requirements, it is designated administrative.

A03 CTS 3.3.3.1 Action c states that the provisions of Specification 3.0.3 are not applicable. CTS 3.0.3 (ITS LCO 3.0.3) does not apply in MODES 5 and 6, which encompasses the condition of during movement of recently irradiated fuel assemblies within the containment. This changes the CTS by eliminating an action requirement that is redundant to other Technical Specification requirements.

The purpose of CTS 3.3.3.1 Action c is to preclude action be taken per Specification 3.0.3 (ITS LCO 3.0.3) to shutdown the unit in the event there are no actions to perform or the actions cannot be performed within the required time when the containment radiation monitoring instrumentation is inoperable. CTS 3.0.3 (ITS LCO 3.0.3) is not applicable while in MODE 5 or 6. Recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours. To access the fuel to begin movement of recently irradiated fuel requires the unit to be in MODE 6. Therefore, the only time recently irradiated fuel can possibly be moved in the containment is following entry into MODE 6 and only up to 72 hours after reaching subcritical conditions. Since LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4, stating that the provisions of Specification 3.0.3 are not applicable is redundant and unnecessary. This change is designated as administrative and is acceptable because it does not result in a technical change to the CTS.

A04 CTS 3.3.3.1 Action b requires, when one or more radiation monitoring channels are inoperable, to take the Action shown in Table 3.3-6. Thus, when the minimum number of required containment radiation monitor channels are inoperable CTS Table 3.3-6, Action 16 (Unit 1) and Action 25 (Unit 2) require compliance with the Action requirements of Specification 3.9.9. ITS 3.3.6

provides explicit actions consistent with the actions of Specification 3.9.9; place and maintain containment purge and exhaust valves in the closed position or suspend movement of recently irradiated fuel assemblies within the containment (per the action of Specification 3.9.3, "Containment Penetrations."). This changes the action presentation of the CTS.

The purpose of the CTS action is to ensure the actions of the supported system, containment isolation valves, are performed consistent with the containment isolation system action requirements specified in CTS 3.9.9. CTS 3.9.9 action requires either suspending all operations involving movement of recently irradiated fuel assemblies within containment or close each of the containment penetrations providing direct access from the containment atmosphere to the outside atmosphere. ITS 3.3.6 Required Action A.1 requires immediately placing and maintaining at least one containment isolation valve in each penetration providing direct access from the containment atmosphere to outside atmosphere in the closed position. ITS 3.3.6 Required Action A.2 requires entering applicable ACTIONS for the affected valves of LCO 3.9.3, "Containment Penetrations," made inoperable by the isolation instrumentation. The applicable ACTIONS in Specification 3.9.3 require immediate suspension of movement of recently irradiated fuel assemblies within containment. This change is designated as administrative and is acceptable because it results in changes in the format and presentation of the CTS requirements.

MORE RESTRICTIVE CHANGES

M01 CTS 3.3.3.1 Action a allows the alarm/trip setpoint of Table 3.3-6 to exceed the setpoint for 4 hours and then to declare the channel inoperable. Upon declaring a required channel inoperable, CTS Table 3.3-6, Action 16 (Unit 1) and Action 25 (Unit 2) require compliance with the Action requirements of Specification 3.9.9. ITS 3.3.6 requires the required channels to be declared inoperable immediately when the actuation setting exceeds the setpoint specified in SR 3.3.6.2 and associated actions must be performed. This changes the CTS by deleting the 4 hour allowance to declare a channel inoperable and requiring the channel to be declared inoperable immediately upon discovery that the channel is outside the required setpoint value.

The purpose of CTS 3.3.3.1 Action a is to provide a short time period to restore an inoperable channel prior to requiring additional action. CTS 3.3.3.1 Action a allows the alarm/trip setpoint of Table 3.3-6 to exceed the setpoint for 4 hours and then to declare the channel inoperable. When LCO 3.3.6 is not met (e.g., SR 3.3.6.2 not met), LCO 3.0.2 requires the associated ACTIONS to be entered with no delay. Each radiation channel provides input to the associated containment high radiation instrument train which actuates the associated containment isolation valves. Therefore, the ITS action is considered appropriate. This change is designated as more restrictive because the Completion Time to declare the instrument channel inoperable following discovery of the channel exceeding it associated trip setpoint has been reduced.

M02 CTS 3.3.3.1 specifies requirements for radiation monitoring instrumentation and list the requirements for each functional unit in CTS Table 3.3-6, including the

containment radiation monitoring instrument channels. ITS 3.3.6 also includes requirements for the containment radiation monitoring instrument channels but also includes requirements for one automatic actuation logic train. LCO 3.3.6 requires, in part, one automatic Actuation Logic train to be OPERABLE and SR 3.3.6.3 requires performance of a CHANNEL FUNCTIONAL TEST on the required Actuation Logic train with a note specifying that testing of the Actuation Logic shall include verification of the proper operation of each initiation relay. The SR Frequency is proposed to be in accordance with the Surveillance Frequency Control Program (SFCP). The proposed actions for the automatic Actuation Logic train are the same as for the containment radiation monitor channels. This changes the CTS by adding requirements in the ITS associated with one train of containment isolation instrument logic.

The purpose of the containment radiation monitoring instrumentation requirements is to ensure isolation capability is maintained to penetrations providing direct access from the containment atmosphere to the outside atmosphere during movement of recently irradiated fuel assemblies within the containment. At least one automatic actuation logic train is required to ensure the applicable containment isolation valves receive an isolation signal from the containment radiation monitoring instrument channels. Therefore, these requirements are appropriate and necessary to ensure the containment radiation monitoring instrumentation can perform its intended safety function. This change is consistent with ISTS 3.3.7.

PSL controls periodic Frequencies for Surveillances in accordance with the SFCP per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, the initial periodic Frequency for ITS SR 3.3.6.3 in accordance with the SFCP will be 31 days consistent with the CHANNEL FUNCTIONAL TEST testing frequency for ISTS SR 3.3.7.3. The periodic Frequency is acceptable based on plant operating experience with regard to channel OPERABILITY, which demonstrates that failure of one automatic actuation logic channel in any 31 day interval is a rare event.

The SFCP was established as described in FPL (PSL Unit 1 and Unit 2) "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program" (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066)

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS 3.3.6, CONTAINMENT ISOLATION INSTRUMENTATION – REFUELING

REMOVED DETAIL CHANGES

LA01 *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-6 for radiation monitoring instrumentation has a column specifying the Measurement Range and a specific measurement range for the containment radiation instrumentation. ITS 3.3.6 does not retain the "MEASUREMENT RANGE" column or the specific measurement range of the containment radiation monitor channels. This changes the CTS by moving the instrument measurement range to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the number of required channels and the appropriate Condition to enter if a required channel becomes inoperable. The low end of the measurement range for the Unit 2 containment radiation monitoring instrumentation is changed from 1 mR/hr to 10 mR/hr to be consistent with the current radiation monitor instrument measurement range. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

PIS (Analog) 3.3.7

Containment Isolation Instrumentation – Refueling

3.3 INSTRUMENTATION (Analog)

Instrumentation – Refueling

3.3.7 Containment Purge Isolation Signal (CPIS) (Analog)

3.3.3.1 LCO 3.3.7 [Four] CPIS containment radiation monitor channels and one CPIS automatic Actuation Logic and one Manual Trip train shall be OPERABLE.

Three

Applicability APPLICABILITY: During movement of [recently] irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One radiation monitor channel inoperable.	A.1 Place the affected channel in trip. OR A.2 Suspend movement of [recently] irradiated fuel assemblies within containment.	4 hours Immediately
B. One required Manual Trip or automatic Actuation Logic train inoperable. OR More than one radiation monitor channel inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1 Place and maintain containment purge and exhaust valves in closed position. OR B.2 Enter applicable Conditions and Required Actions for affected valves of LCO 3.9.3, "Containment Penetrations," made inoperable by isolation instrumentation.	Immediately Immediately

Action b. Table 3.3-6 Function 1.b

at least one containment isolation valve in each penetration providing direct access from containment atmosphere to outside atmosphere

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.3.3.1	SR 3.3.~~7~~.1 (6)	Perform a CHANNEL CHECK on each containment radiation monitor channel. (required)	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3 2 1 2
4.3.3.1 Table 3.3-6 Function 1.b	SR 3.3.~~7~~.2 (6)	Perform a CHANNEL FUNCTIONAL TEST on each containment radiation monitor channel. (required) Verify ~~CPIS~~ high radiation setpoint (containment) is less than or equal to the Allowable Value of ~~[220~~ mR/hr~~]~~. (90)	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3 2 1 2
4.3.3.1	SR 3.3.~~7~~.3 (6)	----------NOTE---------- Testing of Actuation Logic shall include verification of the proper operation of each initiation relay. ---------- Perform a CHANNEL FUNCTIONAL TEST on ~~each CPIS~~ Actuation Logic ~~channel~~. (train) (required)	~~[[31] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3 1 2 1 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
4.3.3.1 SR 3.3.~~7~~.4 (6)	Perform a CHANNEL CALIBRATION on each containment radiation monitor channel. (required)	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3 2 1 2
~~SR 3.3.7.5~~	~~Perform a CHANNEL FUNCTIONAL TEST on each CPIS Manual Trip channel.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~	2 2
~~SR 3.3.7.6~~	~~Verify CPIS response time of each containment radiation channel is within limits.~~	~~[[18] months on a STAGGERED TEST BASIS~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~	2 2

Containment Isolation Instrumentation – Refueling

~~CPIS (Analog)~~ 3.3.~~7~~

3.3 INSTRUMENTATION ~~(Analog)~~

Instrumentation – Refueling

3.3.~~7~~ Containment ~~Purge~~ Isolation ~~Signal (CPIS) (Analog)~~

3.3.3.1 LCO 3.3.~~7~~

Three

~~[Four] CPIS~~ containment radiation monitor channels and one ~~CPIS~~ automatic Actuation Logic ~~and one Manual Trip~~ train shall be OPERABLE.

Applicability APPLICABILITY: During movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One radiation monitor channel inoperable.~~	~~A.1 Place the affected channel in trip.~~ ~~OR~~ ~~A.2 Suspend movement of [recently] irradiated fuel assemblies within containment.~~	~~4 hours~~ ~~Immediately~~
A ~~B~~. One required ~~Manual Trip or~~ automatic Actuation Logic train inoperable. OR One or more required ~~More than one~~ radiation monitor channel<s> inoperable. ~~OR~~ ~~Required Action and associated Completion Time of Condition A not met.~~	~~B~~.1 Place and maintain ~~containment purge and exhaust valves~~ in closed position. OR ~~B~~.2 Enter applicable Conditions and Required Actions for affected valves of LCO 3.9.3, "Containment Penetrations," made inoperable by isolation instrumentation.	Immediately Immediately

Action b. Table 3.3-6 Function 1.b

at least one containment isolation valve in each penetration providing direct access from containment atmosphere to outside atmosphere

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.3.3.1	SR 3.3.~~7~~.1 (6)	Perform a CHANNEL CHECK on each containment radiation monitor channel. (required)	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3 2 1 2
4.3.3.1 Table 3.3-6 Function 1.b	SR 3.3.~~7~~.2 (6)	Perform a CHANNEL FUNCTIONAL TEST on each containment radiation monitor channel. (required) Verify ~~CPIS~~ (containment) high radiation setpoint is less than or equal to the Allowable Value of ~~[220~~ (90) mR/hr~~]~~.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3 2 1 2
4.3.3.1	SR 3.3.~~7~~.3 (6)	----------NOTE---------- Testing of Actuation Logic shall include verification of the proper operation of each initiation relay. ---------- Perform a CHANNEL FUNCTIONAL TEST on ~~each CPIS~~ Actuation Logic ~~channel~~ (train). (required)	~~[[31] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3 1 2 1 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
4.3.3.1 SR 3.3.~~7~~.4 (6)	Perform a CHANNEL CALIBRATION on each containment radiation monitor channel. (required)	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3 2 1) (2)
~~SR 3.3.7.5~~	~~Perform a CHANNEL FUNCTIONAL TEST on each CPIS Manual Trip channel.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~ (2) (2)
~~SR 3.3.7.6~~	~~Verify CPIS response time of each containment radiation channel is within limits.~~	~~[[18] months on a STAGGERED TEST BASIS~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~ (2) (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.6, CONTAINMENT ISOLATION INSTRUMENTATION – REFUELING

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS Specification number and title. Numbering is changed from 3.3.7 to 3.3.6, title is changed to “Containment Isolation Instrumentation – Refueling” and “(Analog)” is deleted from the title.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~7~~ Containment ~~Purge~~ Isolation ~~Signal (CPIS) (Analog)~~

Instrumentation – Refueling

3
3
1

6

BASES

BACKGROUND

This LCO encompasses ~~CPIS actuation, which is a plant specific instrumentation system that performs an actuation Function required for plant protection but is not otherwise included in LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip," or LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS)." This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements, is not included in LCOs 3.3.5 and 3.3.6. Details of this LCO are for illustration only. Individual plants shall include those Functions and LCO requirements applicable to them.~~

Insert 1

containment high radiation instrumentation

The ~~CPIS~~ provides protection from radioactive contamination in the containment in the event a~~n~~ irradiated fuel assembly should be severely damaged during handling.

recently

containment isolation

applicable containment isolation valves close to isolate the containment atmosphere from the outside atmosphere

The ~~CPIS~~ will detect any abnormal amounts of radioactive material in the containment and will initiate ~~purge~~ valve closure to limit the release of radioactivity to the environment. The ~~containment purge supply and exhaust valves are closed on a CPIS~~ when a high radiation level in containment is detected.

CIS logic

logic channels

Insert 2

The ~~CPIS~~ includes two independent, redundant actuation ~~subsystems~~. ~~Where two isolation control valves are provided for a single containment penetration, each valve is controlled by a separate actuation subsystem. Where one valve is available, a single actuation subsystem initiates valve closure. One train also isolates the containment air exhaust fan, whereas the other train actuates the containment air supply fan.~~ A list of actuated valves and an additional description of the ~~CPIS~~ are included in Reference 1. Both trains of ~~CPIS~~ are actuated on a two-out-of-four coincidence from the same four containment radiation sensor~~subsystems~~. ~~Containment purge isolation also occurs on a Containment Isolation Actuation Signal (CIAS). The CIAS is addressed by~~ LCO 3.3.4, "Engineered Safety Features Actuation System (ESFAS) ~~Instrumentation~~."

CIS

s

Insert 3

Logic and Manual Trip

1

Trip Setpoints and Allowable Values

~~Trip setpoints used in the bistables are based on the analytical limits stated in Reference 2.~~ The selection of ~~these~~ trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Value~~s~~

the

1 **INSERT 1**

containment radiation instrumentation which monitors radiation levels in the containment to automatically close at least one containment isolation valve in each penetration that provides direct access from the containment atmosphere to the outside atmosphere following a fuel handling accident (FHA). The containment isolation requirements during refueling are specified in LCO 3.9.3, “Containment Penetrations.”

1 **INSERT 2**

Four channels of containment high radiation instrumentation, with a sensing range of 10 mR/hr to 10^5 R/hr, are provided to the containment isolation signal (CIS) logic. R1

1 **INSERT 3**

Each containment isolation valve is actuated by its associated CIS channel. The instrumentation requirements associated with containment isolation actuation while the unit is operating are addressed in LCO 3.3.3, “Engineered Safety Features Actuation System (ESFAS) Instrumentation,” and

BASES

BACKGROUND (continued)

specified in SR 3.3.~~7~~6.2 ~~are~~ is conservatively adjusted with respect to the ~~analytical~~ design basis limit~~s~~. ~~A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in "Plant Protection System Selection of Trip Setpoint Values" (Ref. 3).~~ The actual nominal trip setpoint entered into the bistable is normally ~~still~~ more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

~~Setpoints in accordance with the Allowable Value will ensure that Safety Limits are not violated during anticipated operational occurrences (AOOs) and the consequences of Design Basis Accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or accident and the equipment functions as designed.~~

Insert 4

APPLICABLE SAFETY ANALYSES

The ~~CPIS~~ satisfies the containment high radiation instrumentation requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Only the Allowable Value~~s are~~ is specified for ~~each~~ the trip Function in the LCO. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that the difference between the nominal trip setpoint and the Allowable Value is equal to or greater than the drift allowance ~~assumed for each trip in the transient and accident analyses~~.

~~Each~~ The Allowable Value specified is more conservative than the ~~analytical~~ design basis limit ~~assumed in the transient and accident analysis~~ in order to account for instrument uncertainties appropriate to the trip Function. ~~These uncertainties are defined in Reference 3.~~ A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

The Bases for the LCO on the ~~CPIS~~ containment high radiation instrumentation are discussed below for each Function:

~~a. Manual Trip~~

~~The LCO on Manual Trip backs up the automatic trips and ensures operators have the capability to rapidly initiate the CPIS Function if any parameter is trending toward its setpoint. At least one channel must be OPERABLE to be consistent with the requirements of LCO 3.9.3, "Containment Penetrations."~~

1

1

INSERT 4

The FHA is a postulated event that involves dropping a single irradiated fuel assembly resulting in damage to the assembly (Ref. 2). The FHA analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. Additionally, containment closure is not assumed for an FHA in the containment. The Technical Requirements Manual includes a decay time requirement that no fuel movement will commence until 72 hours after shutdown. This ensures that the FHA assumptions, including an open containment, are preserved. In the event an FHA in the containment occurs within 72 hours of a unit shutdown, actuation of the CIS when high radiation is indicated in the containment will isolate the containment from the outside environment to minimize the consequences of this improbable event.

BASES

LCO (continued)

a

~~b~~. Containment Radiation - High

Insert 5 → ~~The LCO on the radiation channels requires that all four be OPERABLE.~~

~~[For this unit, the basis for the Containment Radiation - High setpoint is as follows:]~~

2

b

~~c~~. Actuation Logic

Insert 6 → ~~One train of Actuation Logic must be OPERABLE to be consistent with the requirements of LCO 3.9.3. If one fails, it must be restored to OPERABLE status.~~

APPLICABILITY

In MODE 5 or 6, the ~~CPIS isolation of containment purge valves~~ containment high radiation instrumentation is not required to be OPERABLE. However, during movement of ~~[~~recently~~]~~ irradiated fuel ~~[~~(i.e., fuel that has occupied part of a critical reactor core within the previous ~~[X] days)~~]~~, there is the possibility of ~~a fuel handling accident~~ an FHA requiring the ~~CPIS~~ containment high radiation instrumentation on high radiation in containment. Accordingly, the ~~CPIS~~ containment high radiation instrumentation must be OPERABLE during movement of ~~[~~recently~~]~~ irradiated fuel in containment.

72 hours

isolation of penetrations providing direct access from the containment atmosphere to the outside atmosphere

2

2

1

In MODES 1, 2, 3, and 4, ~~the containment purge valves are sealed closed~~.

Insert 7

ACTIONS

A ~~CPIS~~ containment high radiation monitor channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is not large and would result in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it within specification. If the actual trip setpoint is not within the Allowable Value in SR 3.3.~~7~~.2, the channel is inoperable and the appropriate Conditions must be entered.

6

~~In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the sensor, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel should be declared inoperable and the LCO Condition entered for the particular protective function affected.~~

1

INSERT 5

With one Containment Radiation – High channel inoperable, three OPERABLE channels remain available to actuate the CIS logic train concurrent with an additional single failure. Therefore, the LCO requires three of four channels to be OPERABLE.

During refueling conditions, closure of containment isolation valves associated with containment penetrations with direct access from the containment atmosphere to the outside atmosphere on high containment radiation is particularly important since other CIS input parameters are not available (i.e., low pressurizer pressure is bypassed and there is no mechanism for creating a containment pressure transient). With the reactor shutdown, the background radiation seen by the CIS detectors drops by a factor of 10 or more. For these reasons the Containment Radiation - High setpoint requires a much lower setpoint during refueling.

1

INSERT 6

Since LCO 3.9.3 requires, in part, an OPERABLE containment isolation valve for each penetration providing direct access from the containment atmosphere to the outside atmosphere, only one train of Actuation Logic associated with the required containment isolation valves is required to be OPERABLE.

1

INSERT 7

When movement of irradiated fuel assemblies within containment is not being conducted, the potential for an FHA within the containment does not exist. Therefore, due to radioactive decay, an FHA that involves handling non-recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core longer than 72 hours) will result in doses that are well within the guideline values specified in 10 CFR 50.67 even with the containment atmosphere having direct access to the outside atmosphere. Therefore, under these conditions, isolation of the containment penetrations providing direct access to the outside environment, including the associated containment area high radiation instrumentation, is not required to be OPERABLE.

In MODES 1, 2, 3, and 4, LCO 3.3.3 provide instrumentation requirements associated with containment isolation on high containment radiation.

BASES

ACTIONS (continued)

~~When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.~~

~~A.1 and A.2~~

~~Condition A applies to the failure of one Containment Radiation - High CPIS channel. The Required Action is to place the affected channel in the trip condition within 4 hours. The Completion Time accounts for the fact that three redundant channels monitoring containment radiation are still available to provide a single trip input to the CPIS logic to provide the automatic mitigation of a radiation release. Alternately, action must be taken to place the unit in a condition where the LCO does not apply. This does not preclude the movement of fuel to a safe position.~~

A
A
~~B~~.1 and ~~B~~.2
A
or
at least one containment isolation valve in each penetration providing direct access from containment atmosphere to outside atmosphere
or more required
1
3

Condition ~~B~~ applies to the failure of the required ~~Manual Trip or~~ automatic Actuation Logic train, to the failure of ~~more than~~ one radiation monitoring channel~~, or if the Required Action and associated Completion Time of Condition A are not met~~. Required Action ~~B~~.1 is to place ~~the containment purge and exhaust isolation valves~~ in the closed position. The Required Action immediately performs the isolation Function of the ~~CPIS~~. Required Action ~~B~~.2 is to immediately enter the applicable Conditions and Required Actions for the affected isolation valves of LCO 3.9.3, "Containment Penetrations," that were made inoperable by the inoperable instrumentation ~~of the CPIS LCO~~. The Required Action directs the operator to take actions that are appropriate for the containment isolation Function ~~of the CPIS without initiating the containment air supply and exhaust fans~~. The Completion Time accounts for the fact that the automatic capability to isolate containment ~~and initiate supply and exhaust fans~~ on valid containment high radiation signal~~s~~ is degraded during conditions in which a fuel handling accident is possible ~~and CPIS provides the only automatic mitigation of radiation release~~.

s
A
A
a
CIS

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.~~7~~.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

~~[The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SURVEILLANCE REQUIREMENTS (continued)

6
SR 3.3.~~7~~.2 3

A CHANNEL FUNCTIONAL TEST is performed on each containment radiation monitoring channel to ensure the entire channel will perform its intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. ~~This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.~~ There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis. 1

~~[The Frequency of [92] days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any [92] day interval is a rare event.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

6
SR 3.3.~~7~~.3 3

Proper operation of the initiation relays is verified by de-energizing these relays during the CHANNEL FUNCTIONAL TEST of the Actuation Logic. This will actuate the Function, operating ~~all~~ associated equipment. Proper operation of the equipment actuated by ~~each~~ train is thus verified. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. ~~This is acceptable because all of the other required~~ 2

the
the

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.~~ A Note indicates this Surveillance includes verification of operation for each initiation relay. 1

~~[The Frequency of [31] days is based on plant operating experience with regard to channel OPERABILITY, which demonstrates that failure of more than one channel of a given Function in any [31] day interval is a rare event.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

6

SR 3.3.~~7~~.4 3

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

~~[The Frequency is based upon the assumption of an [18] month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

~~SR 3.3.7.5~~

~~A CHANNEL FUNCTIONAL TEST is performed on the manual CPIS actuation circuitry. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.~~ 1

~~This Surveillance verifies that the trip push buttons are capable of opening contacts in the Actuation Logic as designed, de-energizing the initiation relays and providing Manual Trip of the Function. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at a Frequency of once every 18 months.~~ 1 2

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~ 2

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.3.7.6~~

~~This Surveillance ensures that the train actuation response times are less than or equal to the maximum times assumed in the analyses. [The 18 month Frequency is based upon plant operating experience, which shows random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

~~Testing of the final actuating devices, which make up the bulk of the response time, is included. Testing of the final actuating device in one channel is included in the testing of each actuation logic channel.~~

REFERENCES

~~1. FSAR, Section [6.2].~~

~~2~~. FSAR, Section ~~[~~7.3~~]~~.

2. UFSAR, Chapter 15

~~3. "Plant Protection System Selection of Trip Setpoint Values."~~

B 3.3 INSTRUMENTATION ~~(Analog)~~

Instrumentation – Refueling

B 3.3.~~7~~ Containment ~~Purge~~ Isolation ~~Signal (CPIS) (Analog)~~

3
3
1

6

BASES

BACKGROUND

This LCO encompasses ~~CPIS actuation, which is a plant specific instrumentation system that performs an actuation Function required for plant protection but is not otherwise included in LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip," or LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS)." This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements, is not included in LCOs 3.3.5 and 3.3.6. Details of this LCO are for illustration only. Individual plants shall include those Functions and LCO requirements applicable to them.~~

Insert 1

containment high radiation instrumentation

The ~~CPIS~~ provides protection from radioactive contamination in the containment in the event a~~n~~ irradiated fuel assembly should be severely damaged during handling.

recently

containment isolation

applicable containment isolation valves close to isolate the containment atmosphere from the outside atmosphere

The ~~CPIS~~ will detect any abnormal amounts of radioactive material in the containment and will initiate ~~purge~~ valve closure to limit the release of radioactivity to the environment. The ~~containment purge supply and exhaust valves are closed on a CPIS~~ when a high radiation level in containment is detected.

CIAS logic

logic channels

Insert 2

The ~~CPIS~~ includes two independent, redundant actuation ~~subsystems~~. ~~Where two isolation control valves are provided for a single containment penetration, each valve is controlled by a separate actuation subsystem. Where one valve is available, a single actuation subsystem initiates valve closure. One train also isolates the containment air exhaust fan, whereas the other train actuates the containment air supply fan.~~ A list of actuated valves and an additional description of the ~~CPIS~~ are included in Reference 1. Both trains of ~~CPIS~~ are actuated on a two-out-of-four coincidence from the same four containment radiation sensor~~subsystems~~. ~~Containment purge isolation also occurs on a Containment Isolation Actuation Signal (CIAS). The CIAS is addressed by~~ LCO 3.3.4, "Engineered Safety Features Actuation System (ESFAS) ~~Instrumentation~~."

CIAS

s

Insert 3

Logic and Manual Trip

1

Trip Setpoints and Allowable Values

~~Trip setpoints used in the bistables are based on the analytical limits stated in Reference 2.~~ The selection of ~~these~~ trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Value~~s~~

the

1 **INSERT 1**

containment radiation instrumentation which monitors radiation levels in the containment to automatically close at least one containment isolation valve in each penetration that provides direct access from the containment atmosphere to the outside atmosphere following a fuel handling accident (FHA). The containment isolation requirements during refueling are specified in LCO 3.9.3, "Containment Penetrations."

1 **INSERT 2**

Four channels of containment high radiation instrumentation, with a sensing range of 10 mR/hr to 10^7 R/hr, are provided to the containment isolation actuation signal (CIAS) logic. | R1

1 **INSERT 3**

Each containment isolation valve is actuated by its associated CIAS channel. The instrumentation requirements associated with containment isolation actuation while the unit is operating are addressed in LCO 3.3.3, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," and | R1

BASES

BACKGROUND (continued)

6 is

specified in SR 3.3.~~7~~.2 ~~are~~ conservatively adjusted with respect to the ~~analytical~~ limit~~s~~. ~~A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in "Plant Protection System Selection of Trip Setpoint Values" (Ref. 3).~~ The actual nominal trip setpoint entered into the bistable is normally ~~still~~ more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

design basis

~~Setpoints in accordance with the Allowable Value will ensure that Safety Limits are not violated during anticipated operational occurrences (AOOs) and the consequences of Design Basis Accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or accident and the equipment functions as designed.~~

Insert 4

APPLICABLE SAFETY ANALYSES

The ~~CPIS~~ satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

containment high radiation instrumentation

LCO

is the

Only the Allowable Value~~s are~~ specified for ~~each~~ trip Function in the LCO. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that the difference between the nominal trip setpoint and the Allowable Value is equal to or greater than the drift allowance ~~assumed for each trip in the transient and accident analyses~~.

The

design basis

~~Each~~ Allowable Value specified is more conservative than the ~~analytical~~ limit ~~assumed in the transient and accident analysis~~ in order to account for instrument uncertainties appropriate to the trip Function. ~~These uncertainties are defined in Reference 3.~~ A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

containment high radiation instrumentation

The Bases for the LCO on the ~~CPIS~~ are discussed below for each Function:

~~a. Manual Trip~~

~~The LCO on Manual Trip backs up the automatic trips and ensures operators have the capability to rapidly initiate the CPIS Function if any parameter is trending toward its setpoint. At least one channel must be OPERABLE to be consistent with the requirements of LCO 3.9.3, "Containment Penetrations."~~

1

1

INSERT 4

The FHA is a postulated event that involves dropping a single irradiated fuel assembly resulting in damage to the assembly (Ref. 2). The FHA analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. Additionally, containment closure is not assumed for an FHA in the containment. The Technical Requirements Manual includes a decay time requirement that no fuel movement will commence until 72 hours after shutdown. This ensures that the FHA assumptions, including an open containment, are preserved. In the event an FHA in the containment occurs within 72 hours of a unit shutdown, actuation of the CIAS when high radiation is indicated in the containment will isolate the containment from the outside environment to minimize the consequences of this improbable event. | R1

BASES

LCO (continued)

a

~~b~~. Containment Radiation - High

Insert 5 → ~~The LCO on the radiation channels requires that all four be OPERABLE.~~

~~[For this unit, the basis for the Containment Radiation - High setpoint is as follows:]~~

2

b

~~c~~. Actuation Logic

Insert 6 → ~~One train of Actuation Logic must be OPERABLE to be consistent with the requirements of LCO 3.9.3. If one fails, it must be restored to OPERABLE status.~~

APPLICABILITY

containment high radiation instrumentation

In MODE 5 or 6, the ~~CPIS isolation of containment purge valves~~ is not required to be OPERABLE. However, during movement of ~~[~~recently~~]~~ irradiated fuel ~~[~~(i.e., fuel that has occupied part of a critical reactor core within the previous ~~[X] days)~~], there is the possibility of ~~a fuel handling accident~~ requiring the ~~CPIS~~ on high radiation in containment. Accordingly, the ~~CPIS~~ must be OPERABLE during movement of ~~[~~recently~~]~~ irradiated fuel in containment.

72 hours

isolation of penetrations providing direct access from the containment atmosphere to the outside atmosphere

an FHA

containment high radiation instrumentation

2

2

1

Insert 7 → In MODES 1, 2, 3, and 4, ~~the containment purge valves are sealed closed~~.

ACTIONS

containment high radiation monitor

A ~~CPIS~~ channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is not large and would result in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it within specification. If the actual trip setpoint is not within the Allowable Value in SR 3.3.~~7~~.2, the channel is inoperable and the appropriate Conditions must be entered.

6

~~In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the sensor, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel should be declared inoperable and the LCO Condition entered for the particular protective function affected.~~

1

INSERT 5

With one Containment Radiation – High channel inoperable, three OPERABLE channels remain available to actuate the CIAS logic train concurrent with an additional single failure. Therefore, the LCO requires three of four channels to be OPERABLE. | R1

During refueling conditions, closure of containment isolation valves associated with containment penetrations with direct access from the containment atmosphere to the outside atmosphere on high containment radiation is particularly important since other CIAS input parameters are not available (i.e., low pressurizer pressure is bypassed and there is no mechanism for creating a containment pressure transient). With the reactor shutdown, the background radiation seen by the CIAS detectors drops by a factor of 10 or more. For these reasons the Containment Radiation - High setpoint requires a much lower setpoint during refueling. | R1 | R1

1

INSERT 6

Since LCO 3.9.3 requires, in part, an OPERABLE containment isolation valve for each penetration providing direct access from the containment atmosphere to the outside atmosphere, only one train of Actuation Logic associated with the required containment isolation valves is required to be OPERABLE.

1

INSERT 7

When movement of irradiated fuel assemblies within containment is not being conducted, the potential for an FHA within the containment does not exist. Therefore, due to radioactive decay, an FHA that involves handling non-recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core longer than 72 hours) will result in doses that are well within the guideline values specified in 10 CFR 50.67 even with the containment atmosphere having direct access to the outside atmosphere. Therefore, under these conditions, isolation of the containment penetrations providing direct access to the outside environment, including the associated containment area high radiation instrumentation, is not required to be OPERABLE.

In MODES 1, 2, 3, and 4, LCO 3.3.3 provide instrumentation requirements associated with containment isolation on high containment radiation.

BASES

ACTIONS (continued)

~~When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.~~

~~A.1 and A.2~~

~~Condition A applies to the failure of one Containment Radiation - High CPIS channel. The Required Action is to place the affected channel in the trip condition within 4 hours. The Completion Time accounts for the fact that three redundant channels monitoring containment radiation are still available to provide a single trip input to the CPIS logic to provide the automatic mitigation of a radiation release. Alternately, action must be taken to place the unit in a condition where the LCO does not apply. This does not preclude the movement of fuel to a safe position.~~

A A at least one containment isolation valve in each penetration providing direct access from containment atmosphere to outside atmosphere

~~B~~.1 and ~~B~~.2

or

A or more required

Condition ~~B~~ applies to the failure of the required ~~Manual Trip or~~ automatic Actuation Logic train, to the failure of ~~more than~~ one radiation monitoring channel~~, or if the Required Action and associated Completion Time of Condition A are not met~~. Required Action ~~B~~.1 is to place ~~the containment purge and exhaust isolation valves~~ in the closed position. The Required Action immediately performs the isolation Function of the ~~CPIS~~. Required Action ~~B~~.2 is to immediately enter the applicable Conditions and Required Actions for the affected isolation valves of LCO 3.9.3, "Containment Penetrations," that were made inoperable by the inoperable instrumentation ~~of the CPIS LCO~~. The Required Action directs the operator to take actions that are appropriate for the containment isolation Function ~~of the CPIS without initiating the containment air supply and exhaust fans~~. The Completion Time accounts for the fact that the automatic capability to isolate containment ~~and initiate supply and exhaust fans~~ on valid containment high radiation signal~~s~~ is degraded during conditions in which a fuel handling accident is possible ~~and CPIS provides the only automatic mitigation of radiation release~~.

s A A a CIAS

1
3

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.~~7~~.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

~~[The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

6
SR 3.3.~~7~~.2 3

A CHANNEL FUNCTIONAL TEST is performed on each containment radiation monitoring channel to ensure the entire channel will perform its intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. ~~This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.~~ There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis. 1

~~[The Frequency of [92] days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any [92] day interval is a rare event.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 2

6
SR 3.3.~~7~~.3 3

Proper operation of the initiation relays is verified by de-energizing these relays during the CHANNEL FUNCTIONAL TEST of the Actuation Logic. This will actuate the Function, operating ~~all~~ associated equipment. Proper operation of the equipment actuated by ~~each~~ train is thus verified. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. ~~This is acceptable because all of the other required~~ 2

the
the

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.~~ A Note indicates this Surveillance includes verification of operation for each initiation relay. 1

~~[The Frequency of [31] days is based on plant operating experience with regard to channel OPERABILITY, which demonstrates that failure of more than one channel of a given Function in any [31] day interval is a rare event.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

6

SR 3.3.~~7~~.4 3

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

~~[The Frequency is based upon the assumption of an [18] month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

~~SR 3.3.7.5~~

~~A CHANNEL FUNCTIONAL TEST is performed on the manual CPIS actuation circuitry. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.~~ 1

~~This Surveillance verifies that the trip push buttons are capable of opening contacts in the Actuation Logic as designed, de-energizing the initiation relays and providing Manual Trip of the Function. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at a Frequency of once every 18 months.~~ 1 2

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~ 2

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.3.7.6~~

~~This Surveillance ensures that the train actuation response times are less than or equal to the maximum times assumed in the analyses. [The 18 month Frequency is based upon plant operating experience, which shows random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~ 1 2

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~ 2

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 2

~~Testing of the final actuating devices, which make up the bulk of the response time, is included. Testing of the final actuating device in one channel is included in the testing of each actuation logic channel.~~ 1

REFERENCES

~~1. FSAR, Section [6.2].~~
1 U
~~2~~. FSAR, Section ~~[~~7.3~~]~~.
2. UFSAR, Chapter 15
~~3. "Plant Protection System Selection of Trip Setpoint Values."~~

1
1

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.6, BASES, CONTAINMENT ISOLATION INSTRUMENTATION – REFUELING

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS Specification number and title. Numbering is changed from 3.3.7 to 3.3.6, title is changed to “Containment Isolation Instrumentation – Refueling” and “(Analog)” is deleted from the title.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.6, CONTAINMENT ISOLATION INSTRUMENTATION – REFUELING

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

3.3.7, Control Room Isolation Signal (CRIS)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

INSTRUMENTATION

3/4.3.3 ~~MONITORING INSTRUMENTATION~~

CONTROL ROOM ISOLATION SIGNAL (CRIS)

A02

~~RADIATION MONITORING~~

LIMITING CONDITION FOR OPERATION

and one automatic actuation logic train

M02

One CRIS channel per intake

LCO 3.3.7 3.3.3.1 ~~The radiation monitoring instrumentation channels shown in Table 3.3-6~~ shall be OPERABLE ~~with their alarm/trip setpoints within the specified limits~~.

Applicability **APPLICABILITY**: ~~As shown in Table 3.3-6.~~

MODES 1, 2, 3, 4, 5 and 6,
During movement of irradiated fuel assemblies.

ACTION:

~~a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.~~

M01

required CRIS

or one required automatic actuation logic

M02

Condition A

b. With one or more ~~radiation monitoring~~ channels inoperable, ~~take the ACTION shown in Table 3.3-6.~~

A02

~~c. The provisions of Specification 3.0.3 are not applicable.~~

M03

SURVEILLANCE REQUIREMENTS

required CRIS

SR 3.3.7.1
SR 3.3.7.2
SR 3.3.7.4

4.3.3.1 Each ~~radiation monitoring instrumentation~~ channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program.

Add proposed SR 3.3.7.3

M02

SR 3.3.7.5 4.3.3.2 In accordance with the Surveillance Frequency Control Program, ~~each Control Room Isolation radiation monitoring instrumentation~~ channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

required CRIS

~~TABLE 3.3-6~~

~~RADIATION MONITORING INSTRUMENTATION~~

~~INSTRUMENT~~	~~MINIMUM CHANNELS OPERABLE~~	~~APPLICABLE MODES~~	~~ALARM/TRIP SETPOINT~~	~~MEASUREMENT RANGE~~	~~ACTION~~	
~~1. AREA MONITORS~~						
a. Fuel Storage Pool Area	1	*	$\leq$ 15 mR/hr	$10^{-1} – 10^{4}$ mR/hr	13	See ITS 3.3.8
b. Containment (CIS)	3	****	$\leq$ 90 mR/hr	$1 – 10^{5}$ mR/hr	16	See ITS 3.3.6
c. Containment Area – Hi Range	1	1, 2, 3, & 4	$\leq$ 10 R/hr	$1 – 10^{7}$ R/hr	15	See ITS 3.3.9
d. Control Room Isolation ← Signal (CRIS)	1 per intake	ALL MODES and during movement of irradiated fuel assemblies.	$\leq$ 320 cpm	~~$10 – 10^{7}$ cpm~~	~~17~~	LA01, M04
2. PROCESS MONITORS						
a. Containment						
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 – 10^{6}$ cpm	14	See ITS 3.4.15
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 – 10^{6}$ cpm	14	

LCO 3.3.7 Applicability SR 3.3.7.2

* With fuel in the storage pool or building. (See ITS 3.3.8)

**** During movement of recently irradiated fuel assemblies within containment. (See ITS 3.3.6)

~~TABLE 3.3-6 (Continued)~~

~~TABLE NOTATION~~

ACTION 12 - DELETED

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s),and

2) Prepare and submit a Special Report to the Commission pursuant to Specification6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

See ITS 3.3.9

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION A

~~ACTION 17~~ - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION B Note

Add proposed ACTION B

L01

Add proposed ACTION C

M04

INSTRUMENTATION

3/4.3.3 ~~MONITORING INSTRUMENTATION~~

CONTROL ROOM ISOLATION SIGNAL (CRIS)

A02

~~RADIATION MONITORING INSTRUMENTATION~~

LIMITING CONDITION FOR OPERATION

and one automatic actuation logic train

M02

One CRIS channel per intake

LCO 3.3.7 3.3.3.1 ~~The radiation monitoring instrumentation channels shown in Table 3.3-6~~ shall be OPERABLE ~~with their alarm/trip setpoints within the specified limits~~.

Applicability **APPLICABILITY**: ~~As shown in Table 3.3-6.~~

MODES 1, 2, 3, 4, 5 and 6,
During movement of irradiated fuel assemblies.

ACTION:

~~a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.~~

M01

M02

required CRIS

or one required automatic actuation logic

Condition A b. With one or more ~~radiation monitoring~~ channels inoperable, take the ~~ACTION shown in Table 3.3-6.~~

A02

~~c. The provisions of Specification 3.0.3 are not applicable.~~

M03

SURVEILLANCE REQUIREMENTS

required CRIS

SR 3.3.7.1
SR 3.3.7.2
SR 3.3.7.4 4.3.3.1 Each ~~radiation monitoring instrumentation~~ channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program.

Add proposed SR 3.3.7.3

M02

SR 3.3.7.5 4.3.3.2 In accordance with the Surveillance Frequency Control Program, ~~each Control Room Isolation radiation monitoring instrumentation~~ channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

required CRIS

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area					
i. Criticality and Ventilation System Isolation Monitor	4	*	≤ 20 mR/hr	10^{-1} – 10^{4} mR/hr	22
b. Containment Isolation	3	****	≤ 90 mR/hr	1 – 10^{7} mR/hr	25
c. Containment Area – Hi Range	1	1, 2, 3 & 4	Not Applicable	1 - 10^{7} R/hr	27
d. Control Room Isolation ← Signal (CRIS)	1 per intake	ALL MODES and during movement of irradiated fuel assemblies.	≤ 320 cpm	~~10^{-7} – 10^{-2} µCi/cc~~	~~26~~
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10^{-7} – 10^{-2} µCi/cc	23
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 – 10^{7} cpm	23

* With fuel in the storage pool or building.

**** During movement of recently irradiated fuel assemblies within containment.

LCO 3.3.7 Applicability SR 3.3.7.2

LA01

M04

See ITS 3.3.8

See ITS 3.3.6

See ITS 3.3.9

See ITS 3.4.15

See ITS 3.3.8

See ITS 3.3.6

~~TABLE 3.3-6 (Continued)~~

~~ACTION STATEMENTS~~

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION 24 - DELETED

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION A

~~ACTION 26~~ - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION B Note

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

See ITS 3.3.9

Add proposed ACTION B

L01

Add proposed ACTION C

M04

DISCUSSION OF CHANGES
ITS 3.3.7, CONTROL ROOM ISOLATION SIGNAL (CRIS)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.3.3.1 specifies requirements for radiation monitoring instrumentation and list the requirements for each functional unit in CTS Table 3.3-6, including the control room isolation monitoring instrumentation. ITS 3.3.7 provides requirements for the control room isolation signal instrumentation and provides the requirements in the LCO, Applicability, ACTIONS, and Surveillance Requirements. This changes the CTS by presenting the instrument requirements without use of an instrument table. This change is acceptable as it results solely from the change in the format and presentation of the CTS necessary to conform to the ISTS. As the proposed change is the result of changes in the format and presentation of the CTS requirements, it is designated administrative.

MORE RESTRICTIVE CHANGES

M01 CTS 3.3.3.1 Action a allows the alarm/trip setpoint of Table 3.3-6 to exceed the setpoint for 4 hours and then to declare the channel inoperable. Upon declaring a required channel inoperable, CTS Table 3.3-6, Action 17 (Unit 1) and Action 26 (Unit 2) requires placing the CREVS in the recirculation mode of operation. ITS 3.3.6 requires the required channels to be declared inoperable immediately when the actuation setting exceeds the setpoint specified in SR 3.3.6.2 and associated actions must be performed. This changes the CTS by deleting the 4 hour allowance to declare a channel inoperable and requiring the channel to be declared inoperable immediately upon discovery that the channel is outside the required setpoint value.

The purpose of CTS 3.3.3.1 Action a is to provide a short time period to restore an inoperable channel prior to requiring additional action. CTS 3.3.3.1 Action a allows the alarm/trip setpoint of Table 3.3-6 to exceed the setpoint for 4 hours and then to declare the channel inoperable. When LCO 3.3.6 is not met (e.g., SR 3.3.6.2 not met), LCO 3.0.2 requires the associated ACTIONS to be entered with no delay. Each radiation channel provides input to the associated control room ventilation isolation dampers. Therefore, the ITS action is considered appropriate. This change is designated as more restrictive because the Completion Time to declare the instrument channel inoperable following discovery of the channel exceeding it associated trip setpoint has been reduced.

M02 CTS 3.3.3.1 specifies requirements for radiation monitoring instrumentation and list the requirements for each functional unit in CTS Table 3.3-6, including the Control Room outside air intake radiation monitoring instrument channels. ITS 3.3.7 also includes requirements for the control room isolation radiation

monitoring instrument channels but also includes requirements for one automatic actuation logic train. LCO 3.3.7 requires, in part, one automatic Actuation Logic train to be OPERABLE and SR 3.3.7.3 requires performance of a CHANNEL FUNCTIONAL TEST on the required Actuation Logic train with a note specifying that testing of the Actuation Logic shall include verification of the proper operation of each initiation relay. The SR Frequency is proposed to be in accordance with the Surveillance Frequency Control Program (SFCP). This changes the CTS by adding requirements in the ITS associated with one train of control room isolation instrument logic.

The purpose of the control room isolation radiation monitoring instrumentation requirements is to ensure isolation capability is maintained with the ability to isolate the control room atmosphere. At least one automatic actuation logic train is required to ensure the applicable control room ventilation dampers receive an isolation signal from the control room isolation radiation monitoring instrument channels. Therefore, these requirements are appropriate and necessary to ensure the radiation monitoring instrumentation can perform its intended safety function. This change is consistent with ISTS 3.3.8.

PSL controls periodic Frequencies for Surveillances in accordance with the SFCP per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, the initial periodic Frequency for ITS SR 3.3.7.3 in accordance with the SFCP will be 31 days consistent with the CHANNEL FUNCTIONAL TEST testing frequency for ISTS SR 3.3.8.3. The periodic Frequency is acceptable based on plant operating experience with regard to channel OPERABILITY, which demonstrates that failure of one automatic actuation logic channel in any 31 day interval is a rare event.

The SFCP was established as described in FPL (PSL Unit 1 and Unit 2) "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program" (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066).

M03 CTS 3.3.3.1 Action c states that the provisions of Specification 3.0.3 are not applicable. ITS 3.3.7 does not include exceptions to the applicability of LCO 3.0.3. CTS 3.0.3 (ITS LCO 3.0.3) does not apply in MODES 5 and 6, which encompasses the condition of during movement of recently irradiated fuel assemblies within the containment. This changes the CTS by adding the applicability of LCO 3.0.3 when in MODES 1, 2, 3, and 4.

The purpose of the control room isolation function is to ensure a habitable environment for the control room operators. This change is acceptable because LCO 3.0.3 is applicable in MODES 1, 2, 3, and 4 to implement actions when an LCO is not met and either an associated Required Action and Completion is not met with no Condition applicable; or the condition is not specifically addressed by the associated ACTIONS. This change is designated as more restrictive because it adds an additional requirement not in the CTS.

M04 CTS 3.3.3.1 does not explicitly require CRIS to be OPERABLE during movement of irradiated fuel assemblies and does not require the suspension of irradiated fuel assembly movement when control room isolation radiation monitoring channels or actuation logic is inoperable. ITS 3.3.7 Applicability includes the condition, “During movement of irradiated fuel assemblies,” consistent with the CTS 3.7.7.1 (Unit 1) and CTS 3.7.7 (Unit 2) Applicability for the Control Room Emergency Ventilation System. In addition, ITS 3.3.7, Required Action C.2 requires the suspension of movement of irradiated fuel assemblies when required actuation logic or radiation monitors are inoperable during movement of irradiated fuel assemblies if one CREVS train cannot be placed in the recirculation mode. This changes CTS by revising the applicability to include the condition of,” during movement of irradiated fuel assemblies,” and adding the requirement to suspend movement of irradiated fuel assemblies when the control room isolation radiation monitoring actuation logic or monitoring becomes inoperable while in MODES 5 or 6; or, moving irradiated fuel assemblies.

The purpose of the Applicability and action to suspend movement of irradiated fuel assemblies recognizes the fact that the radiation signals are the only Functions available to initiate control room isolation in the event of a fuel handling accident. This change is designated as more restrictive because the change expands the Applicability to include a specified condition and adds a required action that CTS does not have.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-6 for radiation monitoring instrumentation has a column specifying the Measurement Range and a specific measurement range for the control room isolation radiation instrumentation. ITS 3.3.7 does not retain the "MEASUREMENT RANGE" column or the specific measurement range of the radiation monitor channels. This changes the CTS by moving the instrument measurement range to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the number of required channels and the appropriate Condition to enter if a required channel becomes inoperable. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of

detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 - Relaxation of Required Action)* CTS Action b requires when one or more radiation monitoring channels are inoperable to take the ACTION shown in Table 3.3-6. When the minimum number of required control room isolation radiation monitoring channels are inoperable Table 3.3-6 ACTION 17 (Unit 1) and ACTION 26 (Unit 2) require that the control room ventilation system be placed in the recirculation mode of operation within 1 hour. CTS 3.3.3.1 does not specifically address additional required actions if ACTION 17 (Unit 1) and ACTION 26 (Unit 2) cannot meet the required actions. Under similar conditions, ITS 3.3.7 Condition B requires the unit to be in MODE 3 within 6 hours and MODE 4 within 12 hours. Consistent with the ISTS and the CREV Specification ACTIONS, proposed Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. This changes the CTS by allowing a unit shutdown to MODE 4 in lieu of an LCO 3.0.3 shutdown to cold shutdown conditions (MODE 5).

If the required CRIS channel cannot be restored to OPERABLE status, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours (Required Action B.1) and to MODE 4 within 12 hours (Required Action B.2). Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5. In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

This change is acceptable since it provides for an orderly shutdown to hot shutdown condition (MODE 4) if the train is not placed in the recirculation mode within 1 hour and the modification of the end state is consistent with CE-NPSD-1186-A, Revision 00, “Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs,” dated October 2001 (ADAMS Accession No. 110410539). In addition, this allowance is consistent with that provided in the CREVS Specification, when an actual train is inoperable. Once the unit is in MODE 4, the probability of an event requiring the CREVS is minimized. This change is designated as less restrictive because the ITS provides an alternate action than is not currently allowed in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~8~~ Control Room Isolation Signal (CRIS) ~~(Analog)~~

LCO 3.3.3.1 M02

LCO 3.3.~~8~~ One CRIS channel per intake and one automatic logic train shall be OPERABLE.

Applicability

APPLICABILITY: MODES 1, 2, 3, 4, ~~[~~5, and 6~~]~~,
During movement of ~~[recently]~~ irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-6 Action 17 M02 M03 A. One required automatic CRIS ~~Manual Trip~~, Actuation Logic, or ~~[~~one or more required ~~channels of particulate/iodine or gaseous]~~ radiation monitors inoperable in MODE 1, 2, 3, or 4.	A.1 ~~----------NOTE---------- Place Control Room Emergency Air Cleanup System (CREACS) in toxic gas protection mode if automatic transfer to toxic gas protection mode inoperable. ---------- CREVS~~ Place one ~~CREACS~~ train in ~~emergency radiation protection~~ mode. recirculation	1 hour
L01 B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

Table 3.3-6 Action 17

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M04 C. One required CRIS ~~Manual Trip,~~ automatic Actuation Logic, or ~~[one or more required channels of particulate/iodine or gaseous]~~ radiation monitors inoperable ~~[~~in MODE 5 or 6~~]~~, during movement of ~~[recently]~~ irradiated fuel assemblies.	C.1 ~~NOTE Place CREACS in toxic gas protection mode if automatic transfer to toxic gas protection mode inoperable.~~ CREVS Place one ~~CREACS~~ train in ~~emergency radiation protection~~ recirculation mode.	Immediately
	OR	
	C.2.~~1~~ Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately
	~~AND~~	
	~~C.2.2 NOTE Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SDM.~~	
	~~Suspend positive reactivity additions.~~	~~Immediately~~

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.3.3.1	SR 3.3.~~8~~.1 (7)	Perform a CHANNEL CHECK on ~~the~~ (each) required control room radiation monitor channel.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1 2 3
4.3.3.1	SR 3.3.~~8~~.2 (7)	Perform a CHANNEL FUNCTIONAL TEST on the required CRIS radiation monitor channel. Verify CRIS high radiation setpoint is less than or equal to the Allowable Value of ~~[6E4]~~ (320) cpm above normal background.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	2 3

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
M02	SR 3.3.~~8~~.3 7	----------NOTE~~S~~---------- ~~1.~~ Surveillance of Actuation Logic shall include verification of the proper operation of each initiation relay. ~~2. Relays associated with plant equipment that cannot be operated during plant operation are only required to be tested during each MODE 5 entry exceeding 24 hours unless tested within the previous 6 months.~~ ----------		2 1
		Perform a CHANNEL FUNCTIONAL TEST on the required CRIS Actuation Logic channel.	~~[~~[31] days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3
4.3.3.1	SR 3.3.~~8~~.4 7	Perform a CHANNEL CALIBRATION on the required CRIS radiation monitor channel.	~~[~~[18] months ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	2 3

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.3.8.5~~	~~Perform a CHANNEL FUNCTIONAL TEST on the required CRIS Manual Trip channel.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~ 1
4.3.3.2 SR 3.3.~~8.6~~ 7.5	~~[~~ Verify response time of required CRIS channel is within limits.	~~[[18] months~~ 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~ 3

CTS

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~8~~ Control Room Isolation Signal (CRIS) ~~(Analog)~~

LCO 3.3.3.1 M02

LCO 3.3.~~8~~ One CRIS channel per intake and one automatic logic train shall be OPERABLE.

Applicability

APPLICABILITY: MODES 1, 2, 3, 4, ~~[~~5, and 6~~]~~,
During movement of ~~[recently]~~ irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-6 Action 26 M02 M03 A. One required CRIS ~~Manual Trip~~, automatic Actuation Logic, or ~~[~~one or more required ~~channels of particulate/iodine or gaseous]~~ radiation monitors inoperable in MODE 1, 2, 3, or 4.	A.1 ~~----------NOTE----------~~ ~~Place Control Room Emergency Air Cleanup System (CREACS) in toxic gas protection mode if automatic transfer to toxic gas protection mode inoperable.~~ ~~---~~ CREVS Place one ~~CREACS~~ train in ~~emergency radiation protection~~ mode. recirculation	1 hour
L01 B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. --- Table 3.3-6 Action 26 Be in MODE 4.	6 hours 12 hours

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M04 C. One required CRIS ~~Manual Trip,~~ automatic Actuation Logic, or ~~[one or more required channels of particulate/iodine or gaseous]~~ radiation monitors inoperable ~~[~~in MODE 5 or 6~~]~~, during movement of ~~[recently]~~ irradiated fuel assemblies.	C.1 ~~NOTE Place CREACS in toxic gas protection mode if automatic transfer to toxic gas protection mode inoperable.~~ CREVS Place one ~~CREACS~~ train in ~~emergency radiation protection~~ recirculation mode.	Immediately
	OR	
	C.2.~~1~~ Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately
	~~AND~~	
	~~C.2.2 NOTE Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SDM.~~	
	~~Suspend positive reactivity additions.~~	~~Immediately~~

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.3.3.1	SR 3.3.~~8~~.1 (7)	Perform a CHANNEL CHECK on ~~the~~ (each) required control room radiation monitor channel.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1 2 3
4.3.3.1	SR 3.3.~~8~~.2 (7)	Perform a CHANNEL FUNCTIONAL TEST on the required CRIS radiation monitor channel. Verify CRIS high radiation setpoint is less than or equal to the Allowable Value of ~~[6E4]~~ (320) cpm above normal background.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	2 3

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
M02	SR 3.3.~~8~~.3 7	----------NOTE~~S~~---------- ~~1.~~ Surveillance of Actuation Logic shall include verification of the proper operation of each initiation relay. ~~2. Relays associated with plant equipment that cannot be operated during plant operation are only required to be tested during each MODE 5 entry exceeding 24 hours unless tested within the previous 6 months.~~ ----------		2 1
		Perform a CHANNEL FUNCTIONAL TEST on the required CRIS Actuation Logic channel.	~~[~~[31] days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3
4.3.3.1	SR 3.3.~~8~~.4 7	Perform a CHANNEL CALIBRATION on the required CRIS radiation monitor channel.	~~[~~[18] months ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	2 3

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.3.8.5~~	~~Perform a CHANNEL FUNCTIONAL TEST on the required CRIS Manual Trip channel.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~ 1
4.3.3.2 SR 3.3.~~8.6~~ 7.5	~~[~~ Verify response time of required CRIS channel is within limits.	~~[[18] months~~ 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~ 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.7, CONTROL ROOM ISOLATION SIGNAL (CRIS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. Changes are made to reflect the ITS Specification number and title. Numbering is changed from 3.3.8 to 3.3.7 and "(Analog)" is deleted from the title.

3. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

4. Required Action C.2.2 and associated Note are not included in the ITS consistent with the PSL current licensing basis. The boron dilution event analysis for MODES 5 and 6 does not assume operation of the Control Room Emergency Ventilation System or the associated control room isolation signal actuation instrumentation. Therefore, the ISTS Required Action to suspend positive reactivity additions with one CRIS automatic actuation logic or radiation monitor channel inoperable in MODE 5 or 6 is unnecessary. As a result, Required Action C.2.1 is renumbered to C.2.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION ~~(Analog)~~ 2

B 3.3.~~8~~ 7 Control Room Isolation Signal (CRIS) ~~(Analog)~~ 2

BASES

BACKGROUND

This LCO encompasses CRIS actuation, which is a plant specific instrumentation channel that performs an actuation Function required for plant protection but is not otherwise included in LCO 3.3.~~5~~ 4, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual ~~Trip," or LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS)~~." This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements~~,~~ is not included in LCO 3.3.~~5 and LCO 3.3.6~~. ~~Details of this LCO are for illustration only. Individual plants shall include those Functions and LCO requirements that are applicable to them.~~

Actuation

4

6

6

Control Room Emergency Ventilation

The CRIS terminates the normal supply of outside air to the control room and initiates actuation of the ~~Emergency Radiation Protection~~ System to minimize operator radiation exposure. The CRIS includes two independent, redundant subsystems, including actuation trains. Each train employs two separate sensors~~.~~ ~~One sensor detects gaseous activity. The other detects particulate and iodine activity. Since the two sensors detect different types of activity, they are not considered redundant to each other. However, since there are separate sensors in each train, the trains are redundant.~~ If the bistable monitoring either sensor indicates an unsafe condition, that train will be actuated (one-out-of-two logic). The two trains actuate separate equipment. Actuating either train will perform the intended function. Control room isolation also occurs on a ~~Safety Injection Actuation Signal (SIAS)~~.

with a measurement range of 10 to 10^7 cpm

R1

1

Containment Isolation Signal (CIS)

Trip Setpoints and Allowable Values

selection of a setpoint as low as possible without causing spurious alarms or actuations

Trip setpoints used in the bistables are based on the ~~analytical limits~~ (Ref. 1). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Values specified in LCO 3.3.~~8~~ 7 are conservatively adjusted with respect to the ~~analytical limits~~ setpoint. ~~A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in "Plant Protection System Selection of Trip Setpoint Values" (Ref. 2).~~ The actual nominal trip setpoint entered into the bistable is ~~normally~~ still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. ~~If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.~~

2

BASES

BACKGROUND (continued)

Setpoints in accordance with the Allowable Value will ensure that Safety Limits are not violated during anticipated operational occurrences (AOOs) and the consequences of Design Basis Accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or accident and the equipment functions as designed.

APPLICABLE SAFETY ANALYSES

V The CRIS, in conjunction with the Control Room Emergency Ventilation ~~Air Cleanup~~ System (CRE~~AC~~S), maintains the control room atmosphere within conditions suitable for prolonged occupancy throughout the duration of any one of the accidents discussed in Reference ~~1~~. The radiation exposure of control room personnel, through the duration of any one of the postulated accidents discussed in "Accident Analysis," U FSAR, Chapter ~~[~~15~~]~~ (Ref. ~~1~~), does not exceed the limits set by 10 CFR 50, Appendix A, GDC 19 (Ref. 3). 1 3 R1

Loss of Coolant Accident (LOCA) is not a limiting event because the CIS is credited for isolation of the Control Room, no credit is taken for isolation on high radiation in the accident analysis.

The CRIS satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

LCO 3. 3.~~8~~ requires one channel of CRIS to be OPERABLE. The required channel consists of Actuation Logic, ~~Manual Trip,~~ and ~~particulate/iodine and gaseous~~ radiation monitors. The specific Allowable Value~~s~~ for the setpoint~~s~~ of the CRIS ~~are~~ is listed in the SRs.

Only the Allowable Value~~s~~ ~~are~~ is specified for ~~each~~ the trip Function in the LCO. Operation with a trip setpoint less conservative ~~than the nominal trip setpoint,~~ but within its Allowable Value, is acceptable, provided that the difference between the ~~nominal~~ trip setpoint and the Allowable Value is equal to or greater than the drift allowance assumed ~~for each trip~~ in the ~~transient and accident~~ analyses. 1 2

~~Each~~ The Allowable Value specified is more conservative than the ~~analytical limit assumed in the transient and accident analysis~~ setpoint in order to account for instrument uncertainties appropriate to the trip Function. These uncertainties are defined in Reference ~~2~~. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

The Bases for the LCO on the CRIS are discussed below for each Function:

BASES

LCO (continued)

a. ~~Manual Trip~~

~~The LCO on Manual Trip backs up the automatic trips and ensures operators have the capability to rapidly initiate the CRIS Function if any parameter is trending toward its setpoint. One channel must be OPERABLE. This considers that the Manual Trip capability is a backup and that other means are available to actuate the redundant train if required, including manual SIAS.~~

~~b.~~ ~~Airborne~~ Radiation

One ~~Both~~ channel~~s~~ of ~~Airborne~~ Radiation detection in each ~~the~~ required train is ~~are~~ required to be OPERABLE to ensure the control room isolates ~~on either high iodine and high particulate or gaseous concentration~~.

~~[~~For this unit, the basis for the Allowable Value is as follows:~~]~~

b. ~~c.~~ Actuation Logic

A background reading of approximately 40 cpm is typical for the air intake radiation monitors. An actuation setpoint of ≤ 320 cpm is consistent with the EPRI TR-102644 setpoint value recommendation of 8 times background.

One train of Actuation Logic must be OPERABLE, since there are alternate means available to actuate the redundant train, including CIS ~~SIAS~~.

APPLICABILITY

The CRIS Functions must be OPERABLE in MODES 1, 2, 3, 4, ~~[~~5, and 6~~]~~ and during movement of ~~[recently]~~ irradiated fuel assemblies ~~[(i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~ to ensure a habitable environment for the control room operators.

~~REVIEWER'S NOTE~~

~~For those plants that credit gas decay tank rupture accidents, the CRIS must also be OPERABLE in MODES 5 and 6.~~

ACTIONS

A CRIS channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the ~~plant~~ specific setpoint ~~analysis~~. Typically, the drift is not large and would result in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it within specification. If the trip setpoint is not within the Allowable Value, the channel is inoperable and the appropriate Conditions must be entered.

BASES

ACTIONS (continued)

A.1, B.1, B.2, C.1, C.2~~.1,~~ and ~~C.2.2~~ 5

~~Conditions A and C have been modified by a Note, which specifies that CREACS be placed manually in the toxic gas protection mode if the automatic transfer to the toxic gas protection mode is inoperable. [At this unit, the basis for this Note is as follows:]~~

Conditions A, B, and C are applicable to ~~manual and~~ automatic actuation of the CRE~~AC~~S by CRIS. Condition A applies to the failure of the CRIS ~~Manual Trip,~~ Actuation Logic~~, and required particulate/iodine~~ and required ~~gaseous~~ radiation monitor channels in MODE 1, 2, 3, or 4. Entry into this Condition requires action to either restore the failed channel(s) or manually perform the CRIS safety function (Required Action A.1). The Completion Time of 1 hour is sufficient to complete the Required Actions and accounts for the fact that CRIS supplements control room isolation by other Functions (e.g., ~~S~~IAS~~S~~) in MODES 1, 2, 3, and 4. If the channel cannot be restored to OPERABLE status, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours (Required Action B.1) and to MODE 4 within 12 hours (Required Action B.2). 1

CIS

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~4~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 5 1

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

BASES

ACTIONS (continued)

The Completion Times of 6 hours and 12 hours for reaching MODES 3 and 4 from MODE 1 are reasonable, based on operating experience and normal cooldown rates, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant safety systems or operators.

Condition C applies to the failure of CRIS ~~Manual Trip,~~ Actuation Logic~~,~~ and required ~~particulate/iodine and required gaseous~~ radiation monitor ~~channels~~ ~~[~~in MODE 5 or 6~~]~~ or when moving ~~[recently]~~ irradiated assemblies. The Required Actions are immediately taken to place one OPERABLE CRE~~A~~CS train in the ~~emergency radiation protection~~ mode or to suspend ~~positive reactivity additions and~~ movement of ~~[recently]~~ irradiated fuel assemblies. The Completion Time recognizes the fact that the radiation signals are the only Functions available to initiate control room isolation in the event of a fuel handling accident requiring control room isolation. 3 recirculation V 3 1

~~Required Action [C.2.2] is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.~~ 5

SURVEILLANCE REQUIREMENTS

SR 3.3.~~8~~.1 7 2

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

~~At this unit, the following administrative controls and design features (e.g., downscale alarms) immediately alert operations to loss of function in the nonredundant channels.~~ 1

~~[At this unit, verification of sample system alignment and operation for gaseous, particulate, and iodine monitors is required as follows:]~~ 3

SR 3.3.~~8~~.2 7 2

A CHANNEL FUNCTIONAL TEST is performed on the required control room radiation monitoring channel to ensure the entire channel will perform its intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.

SURVEILLANCE REQUIREMENTS (continued)

~~[The Frequency of [92] days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any [92] day interval is a rare event.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 3

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

SR 3.3.~~8~~.3 7 2

Proper operation of the individual initiation relays is verified by de-energizing these relays during the CHANNEL FUNCTIONAL TEST of the Actuation Logic. This will actuate the Function, operating all associated equipment. Proper operation of the equipment actuated by each train is thus verified. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

~~[The Frequency of [31] days is based on plant operating experience with regard to channel OPERABILITY, which demonstrates that failure of more than one channel of a given Function in any [31] days interval is a rare event.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

The Note ~~1~~ indicates this Surveillance includes verification of operation for each initiation relay.

~~Note 2 indicates that relays that cannot be tested at power are excepted from the Surveillance Requirement while at power. These relays must, however, be tested during each entry into MODE 5 exceeding 24 hours unless they have been tested within the previous 6 months.~~

7
SR 3.3.~~8~~.4 2

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

~~[The Frequency is based upon the assumption of an [18] month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.3.8.5~~

~~A CHANNEL FUNCTIONAL TEST is performed on the manual CRIS actuation circuitry. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.~~

~~This test verifies that the trip push buttons are capable of opening contacts in the Actuation Logic as designed, de-energizing the initiation relays and providing Manual Trip of the function. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at a Frequency of once every [18] months.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 5

7.5
~~[~~ SR 3.3.~~8~~.6

This Surveillance ensures that the train actuation response times are less than the maximum times assumed in the analyses. ~~[The [18] month Frequency is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4 3

Testing of the final actuating devices, which make up the bulk of the response time, is included in the Surveillance testing.~~]~~ 3

REFERENCES

1. ~~FSAR, Chapter [15].~~ EPRI Technical Report TR-102644, "Calibration of Radiation Monitors at Nuclear Power Plants." 3
2. ~~"Plant Protection System Selection of Trip Setpoint Values."~~ UFSAR, Chapter 15.
3. 10 CFR 50, Appendix A, GDC 19.
4. PSL-ENG-SEIS-08-017, Control Room Outside Air Intake Radiation Monitor Actuation Setpoint and Response Time Testing Basis Document. 5
~~4~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

1

B 3.3 INSTRUMENTATION ~~(Analog)~~ 2

B 3.3.~~8~~ 7 Control Room Isolation Signal (CRIS) ~~(Analog)~~ 2

BASES

BACKGROUND

This LCO encompasses CRIS actuation, which is a plant specific instrumentation channel that performs an actuation Function required for plant protection but is not otherwise included in LCO 3.3.~~5~~ 4, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual ~~Trip," or LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS)~~." This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements~~,~~ is not included in LCO 3.3.~~5 and LCO 3.3.6~~. ~~Details of this LCO are for illustration only. Individual plants shall include those Functions and LCO requirements that are applicable to them.~~

The CRIS terminates the normal supply of outside air to the control room and initiates actuation of the ~~Emergency Radiation Protection~~ System to minimize operator radiation exposure. The CRIS includes two independent, redundant subsystems, including actuation trains. Each train employs two separate sensors. ~~One sensor detects gaseous activity. The other detects particulate and iodine activity. Since the two sensors detect different types of activity, they are not considered redundant to each other. However, since there are separate sensors in each train, the trains are redundant.~~ If the bistable monitoring either sensor indicates an unsafe condition, that train will be actuated (one-out-of-two logic). The two trains actuate separate equipment. Actuating either train will perform the intended function. Control room isolation also occurs on a ~~Safety Injection Actuation Signal (SIAS)~~.

Trip Setpoints and Allowable Values

Trip setpoints used in the bistables are based on the ~~analytical limits~~ (Ref. 1). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Values specified in LCO 3.3.~~8~~ 7 are conservatively adjusted with respect to the ~~analytical limits~~ setpoint. ~~A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in "Plant Protection System Selection of Trip Setpoint Values" (Ref. 2).~~ The actual nominal trip setpoint entered into the bistable is ~~normally~~ still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. ~~If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.~~

BASES

BACKGROUND (continued)

Setpoints in accordance with the Allowable Value will ensure that Safety Limits are not violated during anticipated operational occurrences (AOOs) and the consequences of Design Basis Accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or accident and the equipment functions as designed.

APPLICABLE SAFETY ANALYSES

The CRIS, in conjunction with the Control Room Emergency ~~Air Cleanup~~ System (CRE~~AC~~S), maintains the control room atmosphere within conditions suitable for prolonged occupancy throughout the duration of any one of the accidents discussed in Reference ~~1~~. The radiation exposure of control room personnel, through the duration of any one of the postulated accidents discussed in "Accident Analysis," FSAR, Chapter ~~[~~15~~]~~ (Ref. ~~1~~), does not exceed the limits set by 10 CFR 50, Appendix A, GDC 19 (Ref. 3).

Loss of Coolant Accident (LOCA) is not a limiting event because the CIAS is credited for isolation of the Control Room, no credit is taken for isolation on high radiation in the accident analysis.

The CRIS satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

LCO 3. 3.~~8~~ requires one channel of CRIS to be OPERABLE. The required channel consists of Actuation Logic, ~~Manual Trip,~~ and ~~particulate/iodine and gaseous~~ radiation monitors. The specific Allowable Value~~s~~ for the setpoint~~s~~ of the CRIS ~~are~~ listed in the SRs.

Only the Allowable Value~~s~~ ~~are~~ specified for ~~each~~ trip Function in the LCO. Operation with a trip setpoint less conservative ~~than the nominal trip setpoint,~~ but within its Allowable Value, is acceptable, provided that the difference between the ~~nominal~~ trip setpoint and the Allowable Value is equal to or greater than the drift allowance assumed ~~for each trip~~ in the ~~transient and accident~~ analyses.

~~E~~ach Allowable Value specified is more conservative than the ~~analytical limit assumed in the transient and accident analysis~~ in order to account for instrument uncertainties appropriate to the trip Function. These uncertainties are defined in Reference ~~2~~. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

The Bases for the LCO on the CRIS are discussed below for each Function:

BASES

LCO (continued)

a. ~~Manual Trip~~

~~The LCO on Manual Trip backs up the automatic trips and ensures operators have the capability to rapidly initiate the CRIS Function if any parameter is trending toward its setpoint. One channel must be OPERABLE. This considers that the Manual Trip capability is a backup and that other means are available to actuate the redundant train if required, including manual SIAS.~~ 1

b. ~~Airborne~~ Radiation

One ~~Both~~ channel~~s~~ of ~~Airborne~~ Radiation detection in each ~~the~~ required train is ~~are~~ required to be OPERABLE to ensure the control room isolates ~~on either high iodine and high particulate or gaseous concentration~~.

~~[~~For this unit, the basis for the Allowable Value is as follows:~~]~~ 3

b. ~~c.~~ Actuation Logic 1

A background reading of approximately 40 cpm is typical for the air intake radiation monitors. An actuation setpoint of ≤ 320 cpm is consistent with the EPRI TR-102644 setpoint value recommendation of 8 times background.

One train of Actuation Logic must be OPERABLE, since there are alternate means available to actuate the redundant train, including CIAS ~~SIAS~~. 1 | R1

APPLICABILITY

The CRIS Functions must be OPERABLE in MODES 1, 2, 3, 4, ~~[~~5, and 6~~]~~ and during movement of ~~[recently]~~ irradiated fuel assemblies ~~[(i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~ to ensure a habitable environment for the control room operators. 3

~~REVIEWER'S NOTE~~

~~For those plants that credit gas decay tank rupture accidents, the CRIS must also be OPERABLE in MODES 5 and 6.~~ 4

ACTIONS

A CRIS channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's function. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the ~~plant~~ specific setpoint ~~analysis~~. Typically, the drift is not large and would result in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it within specification. If the trip setpoint is not within the Allowable Value, the channel is inoperable and the appropriate Conditions must be entered. 1

BASES

ACTIONS (continued)

A.1, B.1, B.2, C.1, C.2~~.1,~~ and ~~C.2.2~~ 5

~~Conditions A and C have been modified by a Note, which specifies that CREACS be placed manually in the toxic gas protection mode if the automatic transfer to the toxic gas protection mode is inoperable. [At this unit, the basis for this Note is as follows:]~~

Conditions A, B, and C are applicable to ~~manual and~~ automatic actuation of the CRE~~AC~~S by CRIS. Condition A applies to the failure of the CRIS ~~Manual Trip,~~ Actuation Logic~~, and required particulate/iodine~~ and required ~~gaseous~~ radiation monitor channels in MODE 1, 2, 3, or 4. Entry into this Condition requires action to either restore the failed channel(s) or manually perform the CRIS safety function (Required Action A.1). The Completion Time of 1 hour is sufficient to complete the Required Actions and accounts for the fact that CRIS supplements control room isolation by other Functions (e.g., ~~S~~IAS) in MODES 1, 2, 3, and 4. If the channel cannot be restored to OPERABLE status, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours (Required Action B.1) and to MODE 4 within 12 hours (Required Action B.2). 1 R1

CIAS

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~4~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 5 1

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

BASES

ACTIONS (continued)

The Completion Times of 6 hours and 12 hours for reaching MODES 3 and 4 from MODE 1 are reasonable, based on operating experience and normal cooldown rates, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant safety systems or operators.

Condition C applies to the failure of CRIS ~~Manual Trip,~~ Actuation Logic~~,~~ and required ~~particulate/iodine and required gaseous~~ radiation monitor ~~channels~~ ~~[~~in MODE 5 or 6~~]~~ or when moving ~~[recently]~~ irradiated assemblies. The Required Actions are immediately taken to place one OPERABLE CRE~~A~~CS train in the ~~emergency radiation protection~~ mode or to suspend ~~positive reactivity additions and~~ movement of ~~[recently]~~ irradiated fuel assemblies. The Completion Time recognizes the fact that the radiation signals are the only Functions available to initiate control room isolation in the event of a fuel handling accident requiring control room isolation.

recirculation
V
3
3
1

~~Required Action [C.2.2] is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.~~

5

SURVEILLANCE REQUIREMENTS

SR 3.3.~~8~~.1 7

2

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

~~At this unit, the following administrative controls and design features (e.g., downscale alarms) immediately alert operations to loss of function in the nonredundant channels.~~ 1

~~[At this unit, verification of sample system alignment and operation for gaseous, particulate, and iodine monitors is required as follows:]~~ 3

SR 3.3.~~8~~.2 7 2

A CHANNEL FUNCTIONAL TEST is performed on the required control room radiation monitoring channel to ensure the entire channel will perform its intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis.

SURVEILLANCE REQUIREMENTS (continued)

~~[The Frequency of [92] days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any [92] day interval is a rare event.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 3

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

SR 3.3.~~8~~.3 7 2

Proper operation of the individual initiation relays is verified by de-energizing these relays during the CHANNEL FUNCTIONAL TEST of the Actuation Logic. This will actuate the Function, operating all associated equipment. Proper operation of the equipment actuated by each train is thus verified. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

~~[The Frequency of [31] days is based on plant operating experience with regard to channel OPERABILITY, which demonstrates that failure of more than one channel of a given Function in any [31] days interval is a rare event.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~
4
3

The Note ~~1~~ indicates this Surveillance includes verification of operation for each initiation relay.

~~Note 2 indicates that relays that cannot be tested at power are excepted from the Surveillance Requirement while at power. These relays must, however, be tested during each entry into MODE 5 exceeding 24 hours unless they have been tested within the previous 6 months.~~

SR 3.3.~~8~~.4 7 2

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

~~[The Frequency is based upon the assumption of an [18] month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~
4
3

~~SR 3.3.8.5~~

~~A CHANNEL FUNCTIONAL TEST is performed on the manual CRIS actuation circuitry. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.~~

~~This test verifies that the trip push buttons are capable of opening contacts in the Actuation Logic as designed, de-energizing the initiation relays and providing Manual Trip of the function. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at a Frequency of once every [18] months.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 5

7.5
~~[~~ SR 3.3.~~8.6~~

This Surveillance ensures that the train actuation response times are less than the maximum times assumed in the analyses. ~~[The [18] month Frequency is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------~~ ~~]~~ 4 3

Testing of the final actuating devices, which make up the bulk of the response time, is included in the Surveillance testing.~~]~~ 3

REFERENCES

1. ~~FSAR, Chapter [15].~~ EPRI Technical Report TR-102644, "Calibration of Radiation Monitors at Nuclear Power Plants." 3
2. ~~"Plant Protection System Selection of Trip Setpoint Values."~~ UFSAR, Chapter 15.
3. 10 CFR 50, Appendix A, GDC 19.
4. PSL-ENG-SEIS-08-017, Control Room Outside Air Intake Radiation Monitor Actuation Setpoint and Response Time Testing Basis Document. 5
~~4~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.7 BASES, CONTROL ROOM ISOLATION SIGNAL (CRIS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. Changes are made to reflect the ITS Specification number and title. Numbering is changed from 3.3.7 to 3.3.6, title is changed to “Containment Isolation Instrumentation – Refueling” and “(Analog)” is deleted from the title.

3. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Changes have been made to reflect changes made to the Specification.

6. Changes have been made to reflect a change to the numbering of ISTS Specification 3.3.5.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.7, CONTROL ROOM ISOLATION SIGNAL (CRIS)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 8

3.3.8, Fuel Pool Area Radiation Instrumentation

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

INSTRUMENTATION

3/4.3.3 ~~MONITORING~~ INSTRUMENTATION

~~RADIATION MONITORING~~

Fuel Pool Area High Radiation Instrumentation

A02

LIMITING CONDITION FOR OPERATION

One fuel pool area high radiation monitoring channel

LCO 3.3.8 3.3.3.1 ~~The radiation monitoring instrumentation channels shown in Table 3.3-6~~ shall be OPERABLE ~~with their alarm/trip setpoints within the specified limits~~.

Applicability **APPLICABILITY**: ~~As shown in Table 3.3-6~~.

During movement of recently irradiated fuel assemblies in the spent fuel storage pool.

L01

ACTION:

~~a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.~~

A02 M01

required

ACTION A

~~b.~~ With one ~~or more radiation monitoring~~ channel~~s~~ inoperable, ~~take the ACTION shown in Table 3.3-6.~~

immediately suspend movement of recently irradiated fuel assemblies in the spent fuel storage pool.

M02

~~c. The provisions of Specification 3.0.3 are not applicable.~~

A03

SURVEILLANCE REQUIREMENTS

required fuel pool area high

SR 3.3.8.1
SR 3.3.8.2
SR 3.3.8.3

4.3.3.1 Each radiation monitoring ~~instrumentation~~ channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program.

4.3.3.2 In accordance with the Surveillance Frequency Control Program, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

See ITS 3.3.7

~~TABLE 3.3-6~~

~~RADIATION MONITORING INSTRUMENTATION~~

A02

~~INSTRUMENT~~	~~MINIMUM CHANNELS OPERABLE~~	~~APPLICABLE MODES~~	ALARM/~~TRIP~~ SETPOINT	~~MEASUREMENT RANGE~~	ACTION
High Radiation Monitoring Instrumentation					
~~1. AREA MONITORS~~					
LCO 3.3.8 SR 3.3.8.2 a. Fuel ~~Storage~~ Pool Area	1	*	≤ 15 mR/hr	~~10^{-1} – 10^{4} mR/hr~~	~~13~~ A LA01
b. Containment (CIS)	3	****	≤ 90 mR/hr	1 – 10^{5} mR/hr	16 See ITS 3.3.6
c. Containment Area – Hi Range	1	1, 2, 3, & 4	≤ 10 R/hr	1 – 10^{7} R/hr	15 See ITS 3.3.9
d. Control Room Isolation	1 per intake	ALL MODES	≤ 320 cpm	10 - 10^{7} cpm	17 See ITS 3.3.7
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 – 10^{6} cpm	14 See ITS 3.4.15
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 – 10^{6} cpm	14

Applicability

* ~~With fuel in the storage pool or building.~~ During movement of recently irradiated fuel assemblies in the spent fuel storage pool.

**** During movement of recently irradiated fuel assemblies within containment. See ITS 3.3.6

L01

~~TABLE 3.3-6 (Continued)~~

~~TABLE NOTATION~~

~~ACTION 12 - DELETED~~

Immediately suspend movement of recently irradiated fuel assemblies in the spent fuel storage pool.

M02

~~ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.~~

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s),and

2) Prepare and submit a Special Report to the Commission pursuant to Specification6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

See ITS 3.3.9

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

INSTRUMENTATION

3/4.3.3 ~~MONITORING~~ INSTRUMENTATION

Fuel Pool Area High Radiation Instrumentation

A02

~~RADIATION MONITORING INSTRUMENTATION~~

LIMITING CONDITION FOR OPERATION

LCO 3.3.8

3.3.3.1 ~~The radiation monitoring instrumentation channels shown in Table 3.3-6~~ shall be OPERABLE ~~with their alarm/trip setpoints within the specified limits~~.

Two fuel pool area high radiation instrument trains

Applicability

APPLICABILITY: ~~As shown in Table 3.3-6~~.

During movement of recently irradiated fuel assemblies in the spent fuel storage pool.

L01

ACTION:

~~a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.~~

A02 M01

Condition A

instrument train

~~b.~~ With one ~~or more radiation monitoring channels~~ inoperable, ~~take the ACTION shown in Table 3.3-6~~.

restore train to OPERABLE status within 7 days

M02

~~c. The provisions of Specification 3.0.3 are not applicable.~~

A03 M02

Add proposed ACTION B

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1
SR 3.3.8.2
SR 3.3.8.3

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program.

4.3.3.2 In accordance with the Surveillance Frequency Control Program, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

See ITS 3.3.7

~~TABLE 3.3-6~~

~~RADIATION MONITORING INSTRUMENTATION~~

~~INSTRUMENT~~	~~MINIMUM~~ CHANNELS OPERABLE	~~APPLICABLE MODES~~	~~ALARM/~~TRIP SETPOINT	~~MEASUREMENT RANGE~~	ACTION
~~1. AREA MONITORS~~					
LCO 3.3.8 a. Fuel ~~Storage~~ Pool Area					
SR 3.3.8.3 i. Criticality and Ventilation System Isolation Monitor	~~4~~ 2 per instrument logic	*	≤ 20 mR/hr	~~10^{-1} – 10^{4} mR/hr~~	~~22~~ A, B
b. Containment Isolation	3	****	≤ 90 mR/hr	1 – 10^{7} mR/hr	25
c. Containment Area – Hi Range	1	1, 2, 3 & 4	Not Applicable	1 - 10^{7} R/hr	27
d. Control Room Isolation	1 per intake	ALL MODES	≤ 320 cpm	10^{-7} – 10^{-2} µCi/cc	26
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10^{-7} – 10^{-2} µCi/cc	23
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 – 10^{7} cpm	23

* ~~With fuel in the storage pool or building~~ During movement of recently irradiated fuel assemblies in the spent fuel storage pool.

Applicability **** During movement of recently irradiated fuel assemblies within containment.

~~TABLE 3.3-6 (Continued)~~

~~ACTION STATEMENTS~~

~~ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.~~

Add proposed ACTIONS A and B

M02

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION 24 - DELETED

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION A

See ITS 3.3.6

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

See ITS 3.3.9

DISCUSSION OF CHANGES
ITS 3.3.8, FUEL POOL AREA HIGH RADIATION INSTRUMENTATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.3.3.1 specify requirements for radiation monitoring instrumentation and list the requirements for each functional unit in CTS Table 3.3-6, including the Fuel Pool Storage Area Monitor instrumentation. ITS 3.3.8 provides requirements for the fuel pool area radiation instrumentation and provides the requirements in the LCO, Applicability, ACTIONS, and Surveillance Requirements. This changes the CTS by presenting the instrument requirements without use of an instrument table. This change is acceptable as it results solely from the change in the format and presentation of the CTS necessary to conform to the ISTS. This change is designated as administrative and is acceptable because it is the result of changes in the format and presentation of the CTS requirements and does not result in a technical change to the CTS.

A03 CTS 3.3.3.1 Action c states that the provisions of Specification 3.0.3 are not applicable. CTS 3.0.3 (ITS LCO 3.0.3) does not apply in MODES 5 and 6, which encompasses the condition of during movement of recently irradiated fuel assemblies. This changes the CTS by eliminating an action requirement that is redundant to other Technical Specification requirements.

The purpose of CTS 3.3.3.1 Action c is to preclude action be taken per Specification 3.0.3 (ITS LCO 3.0.3) to shutdown the unit in the event there are no actions to perform or the actions cannot be performed within the required time when the fuel pool area radiation monitoring instrumentation is inoperable. The ITS Applicability is changed to “During movement of recently irradiated fuel assemblies in the spent fuel storage pool (Refer to Discussion of Change L01). CTS 3.0.3 (ITS LCO 3.0.3) is not applicable while in MODE 5 or 6. Recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours. To access the fuel to begin movement of recently irradiated fuel requires the unit to be in MODE 6. Therefore, the only time recently irradiated fuel can possibly be moved in the spent fuel storage pool is following entry into MODE 6 and only up to 72 hours after reaching subcritical conditions. Since LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4, stating that the provisions of Specification 3.0.3 are not applicable is redundant and unnecessary. This change is designated as administrative and is acceptable because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.3.3.1 Action a allows the alarm/trip setpoint of Table 3.3-6 to exceed the setpoint for 4 hours and then to declare the channel inoperable. Upon declaring the channel inoperable, CTS Table 3.3-6, Action 13 (Unit 1) and Action 22 (Unit 2) require area surveys of the monitored area to be performed with portable monitoring instrumentation at least one every 24 hours. ITS 3.3.8 requires the channel to be declared inoperable immediately when the alarm (Unit 1) / actuation (Unit 2) setting exceeds the setpoint specified in SR 3.3.8.3 and associated actions must be performed. This changes the CTS by deleting the 4 hour allowance to declare a channel inoperable and requiring the channel to be declared inoperable immediately upon discovery that the channel is outside the required trip/alarm setpoint value.

The purpose of CTS 3.3.3.1 Action a is to provide a short time period to restore an inoperable channel prior to requiring additional action. CTS 3.3.3.1 Action a allows the alarm/trip setpoint of Table 3.3-6 to exceed the setpoint for 4 hours and then to declare the channel inoperable. When LCO 3.3.8 is not met (i.e., SR 3.3.8.3 not met), LCO 3.0.2 requires the associated ACTIONS to be entered with no delay. For Unit 2, each radiation channel provides input to the associated fuel pool area high radiation instrument train which actuates the associated Shield Building Ventilation System (SBVS) train. Therefore, the ITS action is considered appropriate. This change is designated as more restrictive because the Completion Time to declare the required fuel pool area radiation instrument channels inoperable following discovery of an associated channel exceeding it associated trip setpoint has been reduced.

M02 CTS 3.3.3.1 Action b requires, when one or more radiation monitoring channels are inoperable, to take the ACTION shown in Table 3.3-6. Thus, when the Unit 1 Fuel Storage Pool Area monitor channel is inoperable CTS Table 3.3-6, Action 13 requires area surveys of the monitored area to be performed with portable monitoring instrumentation at least once every 24 hours. When one or more Unit 2 Fuel Storage Pool Area Criticality and Ventilation System Isolation Monitor channels are inoperable CTS Table 3.3-6, Action 22 also requires area surveys of the monitored area to be performed with portable monitoring instrumentation at least once every 24 hours. Unit 1 ITS 3.3.8 ACTIONS require, when the required fuel pool area high radiation monitoring channel is inoperable, immediate suspension of movement of recently irradiated fuel assemblies within the spent fuel storage pool. Unit 2 ITS 3.3.8 ACTION A requires, with one fuel pool area high radiation instrument train inoperable, the associated fuel pool area high radiation instrument train to be restored to OPERABLE status within 7 days. With two fuel pool area high radiation instrument trains inoperable or if one inoperable instrument train cannot be restored within 7 days, Unit 2 ITS 3.3.8 ACTION B requires immediate suspension of recently irradiated fuel assemblies within the spent fuel storage pool (i.e., exit the Applicability). This changes the CTS actions by replacing the requirement to perform radiation surveys every 24 hours with portable monitoring instrumentation with actions to restore the instrumentation to OPERABLE status (Unit 2) or exit the Applicability (Unit 1 and Unit 2).

DISCUSSION OF CHANGES
ITS 3.3.8, FUEL POOL AREA HIGH RADIATION INSTRUMENTATION

The purpose of the CTS 3.3.3.1 requirements is to ensure radiation instrumentation required to actuate safety related systems are OPERABLE. The fuel storage pool area monitoring instrumentation for Unit 1 is provided to monitor radiation levels in the fuel pool area to alert the operator of the need to filter airborne radioactive particulate following a fuel handling accident (FHA). On a Unit 1 high fuel pool area radiation signal, the Fuel Pool Area Ventilation System is manually started and aligned to exhaust the spent fuel storage pool area through the Fuel Pool Area Ventilation System filtration unit, if the alarm is due to an FHA. The fuel storage pool area monitoring instrumentation for Unit 2 is provided to actuate the SBVS and aligns the system to the area of the spent fuel storage pool to filter airborne radioactive particulate following an FHA. On a Unit 2 high fuel pool area radiation signal, the SBVS automatically starts and aligns to exhaust the spent fuel storage pool area through the SBVS filtration units.

10 CFR 50.36(c)(2)(i) requires, in part, when LCO is not met, the licensee to shut down the reactor or follow any remedial action permitted by the Technical Specifications until the condition can be met. Although the CTS action to perform area surveys of the monitored area with portable monitoring instrumentation at least once every 24 hours does provide compensatory actions for monitoring radiation levels during normal operation, the action requirement does not provide an appropriate remedial action necessary to place the plant in a safe condition with required monitoring instrumentation inoperable that is necessary to support the mitigation of an FHA in the spent fuel storage pool. Unit 1 ITS 3.3.8 ACTION A is acceptable because the action places the unit in a condition in which the LCO does not apply (i.e., immediately suspend movement of recently irradiated fuel assemblies in the spent fuel storage pool). This does not preclude the movement of fuel to a safe position. Unit 2 ITS 3.3.8 ACTION A is acceptable because when one fuel pool area high radiation instrument train is inoperable (e.g., one or two required fuel pool area high radiation channels are inoperable or an actuation logic relay is inoperable), the inoperable instrument train must be restored to OPERABLE status within 7 days. The remaining OPERABLE instrument train in this degraded condition is capable of actuating one SBVS train following an FHA. The Completion Time is consistent with the Completion Time to restore one inoperable SBVS train to OPERABLE status in the ACTIONS of Specification 3.6.9, “Shield Building Ventilation System (SBVS),” and is based on consideration of such factors as the availability of the OPERABLE redundant instrument train and the low probability of an FHA occurring during this period. Unit 2 ITS 3.3.8 ACTION B is acceptable because, when one fuel pool area high radiation instrument train cannot be restored to OPERABLE status within 7 days or when two fuel pool area high radiation instrument trains are inoperable during movement of recently irradiated fuel assemblies, the action places the unit in a condition in which the LCO does not apply (i.e., immediately suspend movement of recently irradiated fuel assemblies in the spent fuel storage pool). This does not preclude the movement of fuel to a safe position. This change is designated as more restrictive because the current actions that allow operation to continue for an unlimited period of time with the inoperable radiation monitoring instrumentation have been replaced with actions that limit the time operation in the specified condition can continue with the inoperable instrumentation.

DISCUSSION OF CHANGES
ITS 3.3.8, FUEL POOL AREA HIGH RADIATION INSTRUMENTATION

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-6 for radiation monitoring instrumentation has a column specifying the Measurement Range and a specific measurement range for the fuel storage pool area instrumentation. ITS 3.3.8 does not retain the "MEASUREMENT RANGE" column or the specific measurement range of the fuel pool area high radiation instrumentation. This changes the CTS by moving the instrument measurement range to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the number of required channels and the appropriate Condition to enter if a required channel becomes inoperable. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 (*Category 2 – Relaxation of Applicability*) CTS 3.3.3.1 requires the radiation monitoring instrumentation channels shown in Table 3.3-6 to be OPERABLE, which includes the Unit 1 Fuel Storage Pool Area Monitors (Instrument 1.a) and the Unit 2 Fuel Storage Pool Area Criticality and Ventilation System Isolation Monitors (Instrument 1.a.i). Footnote * specifies the Applicability for Unit 1 Instrument 1.a and Unit 2 Instrument 1.a.i and states," With fuel in the storage pool or building." The ITS 3.3.8 Applicability requires the instrumentation to be OPERABLE, "During movement of recently irradiated fuel assemblies in the spent fuel storage pool." This changes the CTS by relaxing the Applicability of the Fuel Storage Pool Area Monitors to only when there is a potential for a fuel handling accident (FHA) involving recently irradiated fuel, that is, during the movement of recently irradiated fuel assemblies in the spent fuel storage pool. | R1

The purpose of the area radiation monitoring instrumentation is to ensure that the Fuel Pool Ventilation System (Unit 1) or SBVS (Unit 2) can provide filtration to minimize the consequences of an FHA involving recently irradiated fuel (i.e., an irradiated fuel assembly that has occupied part of a critical reactor core within the previous 72 hours). This change is acceptable because the requirements continue to ensure that the structures, systems, and components are maintained in the specified conditions needed to minimize event consequences during an

FHA involving recently irradiated fuel. The FHA analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. It is improbable to move irradiated fuel within 72 hours of a plant shutdown (i.e., keff < 0.99) because of the physical time required to perform a controlled plant shutdown, cooldown and depressurize the Reactor Coolant System (RCS), and disassemble the reactor vessel to access irradiated fuel to begin fuel movement. Additionally, filtration of the fuel handling building is not assumed for an FHA in the spent fuel storage pool. The ITS requires OPERABLE fuel pool area high radiation instrumentation to ensure that, in the unlikely event an FHA in the spent fuel storage pool occurs within 72 hours of a unit shutdown, fuel pool area ventilation and filtration are provided to minimize the consequences of this improbable event. This change is designated as less restrictive because the LCO requirements are applicable in fewer conditions than in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

~~SBFAS (Analog)~~ Fuel Pool Area Radiation Instrumentation

3.3.~~10~~ 8 1 3

3.3 INSTRUMENTATION ~~(Analog)~~

Fuel Pool Area Radiation Instrumentation 1 3

3.3.~~10~~ 8 ~~Shield Building Filtration Actuation Signal (SBFAS) (Analog)~~ 3

One fuel pool area high radiation monitoring channel

3.3.3.1 Table 3.3-6 Function 1.a

LCO 3.3.~~10~~ ~~Two channels of SBFAS automatic and two channels of Manual Trip~~ shall be OPERABLE. 3 1

Applicability

APPLICABILITY: ~~MODES 1, 2, 3, and 4~~. During movement of recently irradiated fuel assemblies in the spent fuel storage pool 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b. DOC M02 A. One required ~~Manual Trip or Actuation Logic~~ channel inoperable.	A.1 ~~Restore the channel to OPERABLE status.~~	~~48 hours~~ Immediately 1
~~B. Required Action and associated Completion Time not met.~~	~~B.1~~ ~~Be in MODE 3.~~ ~~AND~~ ~~B.2~~ ~~----------NOTE----------~~ ~~LCO 3.0.4.a is not applicable when entering MODE 4.~~ ~~----------~~ ~~Be in MODE 4~~.	~~6 hours~~ Suspend movement of recently irradiated fuel assemblies in the spent fuel storage pool. 1 ~~12 hours~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.3.3.1	SR 3.3.~~10~~.1 (8)	Perform a CHANNEL ~~FUNCTIONAL TEST~~ (CHECK) on ~~each SBFAS automatic actuation~~ channel. (required fuel pool area high radiation monitoring)	~~[~~ ~~[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3 2 1 2
4.3.3.1	SR 3.3.~~10~~.2 (8)	Perform a CHANNEL FUNCTIONAL TEST on ~~each SBFAS Manual Trip~~ channel. (required fuel pool area high radiation monitoring)	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3 2 1 2

Insert 1

1

1

INSERT 1

4.3.3.1	SR 3.3.8.3	Perform a CHANNEL CALIBRATION on required fuel pool area high radiation monitoring channel. Verify alarm setting is less than or equal to the setpoint of 15 mR/hr.	In accordance with the Surveillance Frequency Control Program

CTS

~~SBFAS (Analog)~~ 3.3.~~10~~

Fuel Pool Area Radiation Instrumentation

8

3.3 INSTRUMENTATION ~~(Analog)~~

Fuel Pool Area Radiation Instrumentation

3.3.~~10~~ ~~Shield Building Filtration Actuation Signal (SBFAS) (Analog)~~

8

fuel pool area high radiation instrument trains

3.3.3.1 Table 3.3-6 Function 1.a

LCO 3.3.~~10~~ Two ~~channels of SBFAS automatic and two channels of Manual Trip~~ shall be OPERABLE.

Applicability

APPLICABILITY: ~~MODES 1, 2, 3, and 4~~.

During movement of recently irradiated fuel assemblies in the spent fuel storage pool

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b. DOC M02 A. One ~~Manual Trip or Actuation Logic channel~~ inoperable. (instrument train)	A.1 Restore ~~the channel~~ (instrument train) to OPERABLE status.	~~48 hours~~ 7 days
B. Required Action and associated Completion Time not met. OR Two instrument trains inoperable. Table 3.3-6 Function 1.a DOC M02	B.1 ~~Be in MODE 3.~~ ~~AND~~ ~~B.2~~ ~~NOTE LCO 3.0.4.a is not applicable when entering MODE 4.~~ ~~Be in MODE 4~~.	~~6 hours~~ Immediately Suspend movement of recently irradiated fuel assemblies in the spent fuel storage pool. ~~12 hours~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.3.3.1	SR 3.3.~~10~~.1 Perform a CHANNEL ~~FUNCTIONAL TEST~~ on each ~~SBFAS automatic actuation~~ channel.	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
4.3.3.1	SR 3.3.~~10~~. 2 Perform a CHANNEL FUNCTIONAL TEST on each ~~SBFAS Manual Trip channel~~.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

1

INSERT 1

4.3.3.1	SR 3.3.8.3	Perform a CHANNEL CALIBRATION on each fuel pool area high radiation instrument train. Verify actuation setting is less than or equal to the setpoint of 20 mR/hr.	In accordance with the Surveillance Frequency Control Program

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.8, FUEL POOL AREA RADIATION INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS Specification number and title. Numbering is changed from 3.3.10 to 3.3.8 and “(Analog)” is deleted from the title.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUCTIONS (Analog)

B 3.3.10 Shield Building Filtration Actuation Signal (SBFAS) (Analog)

8

Fuel Pool Area Radiation Instrumentation

3

1

BASES

BACKGROUND

This LCO encompasses the SBFAS, which is a plant specific instrumentation system that performs an actuation Function required for plant protection but is not otherwise included in LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip," or LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS)." This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements, is not included in LCOs 3.3.5 and 3.3.6. Details of this LCO are for illustration only. Individual plants shall include those Functions and LCO requirements that are applicable to them.

Insert 1

1

APPLICABLE SAFETY ANALYSES

monitor

The SBFAS is required to filter the air space between the containment and shield building during a loss of coolant accident (LOCA), as discussed in FSAR, Chapter 15 (Ref. 1).

Insert 2

1

fuel pool area radiation instrumentation

The SBFAS satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

1

LCO

The LCO on equipment OPERABILITY ensures that the SBFAS will perform as required when called upon.

fuel pool area radiation instrumentation

1

one fuel pool area high radiation monitoring channel

The LCO requires two channels of SBFAS automatic and Manual Trip to be OPERABLE. Two channels are necessary to ensure the required redundancy should one channel become inoperable.

Insert 3

1

APPLICABILITY

The SBFAS must be OPERABLE in MODES 1, 2, 3, and 4, since the possibility of a LOCA is greatest in these MODES. In MODE 5 or 6 the probability of a LOCA is greatly diminished, and there is ample time to respond manually to a LOCA event.

Insert 4

1

ACTIONS

When the number of inoperable channels in a trip Function exceeds those specified in the Conditions associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

1

1 **INSERT 1**

fuel pool area radiation instrumentation which monitors radiation levels in the fuel pool area to alert the operator of the need to filter airborne radioactive particulate following a fuel handling accident (FHA). The Fuel Pool Area Ventilation System requirements are specified in LCO 3.7.17, "Fuel Pool Area Ventilation System." On a high fuel pool area radiation signal, the Fuel Pool Area Ventilation System is manually started and aligned to exhaust the spent fuel storage pool area through the Fuel Pool Area Ventilation System filtration unit if the alarm is due to an FHA. In this alignment, the system achieves and maintains a negative air pressure in the fuel handling building. Two channels of fuel pool area high radiation instrumentation, with a sensing range of 1 to 10E4 mR/hr, are provided to monitor the fuel pool area. | R1

1 **INSERT 2**

spent fuel storage pool area during an FHA involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 72 hours). The analysis of the FHA is given in Reference 1

The FHA analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. Additionally, filtration of the fuel handling building is not assumed for an FHA in the spent fuel storage pool. The Technical Requirements Manual includes a decay time requirement that no fuel movement will commence until 72 hours after shutdown. This ensures that the FHA assumptions, including no filtration of the fuel handling building, are preserved. In the event an FHA in the spent fuel storage pool occurs within 72 hours of a unit shutdown, actuation of the Fuel Pool Area Ventilation System when high radiation is indicated in the fuel pool area high will provide filtration to minimize the consequences of this improbable event.

1 **INSERT 3**

Since the Fuel Pool Area Ventilation System consists of a common filtration unit and is a manually actuated system, only one channel of the fuel pool area radiation monitoring instrumentation is required to be OPERABLE.

1

INSERT 4

During movement of recently irradiated fuel assemblies in the spent fuel storage pool, the fuel pool area ventilation high radiation instrumentation is required to be OPERABLE to support the Fuel Pool Area Ventilation System. Due to radioactive decay associated with irradiated fuel assemblies in the spent fuel storage pool, the Fuel Pool Area Ventilation System is only required to mitigate FHAs that involve handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 72 hours).

When movement of irradiated fuel assemblies within the spent fuel storage pool is not being conducted, the potential for an FHA within the fuel pool area does not exist. Additionally, due to radioactive decay, an FHA that involves handling non-recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core longer than 72 hours) will result in doses that are well within the guideline values specified in 10 CFR 50.67 even without fuel pool area ventilation filtration capability. Therefore, under these conditions the Fuel Pool Area Ventilation System, including the associated high radiation monitoring instrumentation, is not required to be OPERABLE.

BASES

ACTIONS (continued)

A.1

Insert 5

Condition A applies to the failure of one SBFAS Manual Trip channel or of one Actuation Logic associated with the Chemical and Volume Control System Isolation Signal or SBFAS. Required Action A.1 requires restoration of the inoperable channel to restore redundancy of the affected Function. The Completion Time of 48 hours is consistent with the Completion Time of other ESFAS Functions employing similar logic and should be adequate for most repairs while minimizing the risk of operating with an inoperable channel for a manually actuated Function. 1

B.1 and B.2

REVIEWER'S NOTE

Adoption of a MODE 4 end state requires the licensee to make the following commitments:

1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F]. 2

2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010. 2

Condition B specifies the shutdown track to be followed if the Required Action and associated Completion Time of Condition A are not met. If Required Action A.1 cannot be met within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. Inoperability of the SBFAS has no effect on plant core damage frequency in any operational MODE. 1

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 1

1

INSERT 5

When the required fuel pool area ventilation high radiation monitoring channel is inoperable during movement of recently irradiated fuel assemblies, action must be taken to place the unit in a condition in which the LCO does not apply. This involves immediately suspending movement of recently irradiated fuel assemblies in the spent fuel storage pool. This does not preclude the movement of fuel to a safe position.

BASES

ACTIONS (continued)

~~Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~

1

~~The Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.~~

1

SURVEILLANCE REQUIREMENTS

SR 3.3.~~10~~.1

8

3

Insert 6

~~The SBFAS can be initiated either on a Safety Injection Actuation Signal (SIAS) or manually. This Surveillance is a restatement of SR 3.3.5.1 on the SIAS Function. Performing SR 3.3.5.1 satisfies this Surveillance. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. [The Frequency is the same as that for SR 3.3.5.1.~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~- - - - - - - - - - - - - - - - - REVIEWER'S NOTE - - - - - - - - - - - - - - - - -~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~- - - - - - - - - - - - - - - - -

1

INSERT 6

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.~~10~~.2

8

fuel pool area high radiation instrumentation

3
1

A CHANNEL FUNCTIONAL TEST is performed on the ~~manual SBFAS~~ actuation circuitry. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. ~~This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions~~.

1

Insert 7

~~This Surveillance verifies that the trip push buttons are capable of opening contacts in the Actuation Logic as designed, de-energizing the initiation relays and providing Manual Trip of the Function. [The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at a Frequency of once every [18] months~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

2

Insert 8

~~----------]~~

1

REFERENCES

1. FSAR, Chapter ~~[~~15~~]~~.

U

1

~~2. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.~~

1

UFSAR, Section 7.3

1

INSERT 7

The CHANNEL FUNCTIONAL TEST tests the individual sensor subsystems using an analog test input to each bistable. A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified tolerance around the setpoint. Any setpoint adjustment shall be consistent with the assumptions of the plant specific setpoint analysis.

1

INSERT 8

SR 3.3.8.3 CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the extension analysis. The requirements for this review are outlined in Reference 2.

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~10~~ ~~Shield Building Filtration Actuation Signal (SBFAS) (Analog)~~

8

Fuel Pool Area Radiation Instrumentation

BASES

BACKGROUND

This LCO encompasses the ~~SBFAS, which is a plant specific instrumentation system that performs an actuation Function required for plant protection but is not otherwise included in LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip," or LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS)." This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements, is not included in LCOs 3.3.5 and 3.3.6. Details of this LCO are for illustration only. Individual plants shall include those Functions and LCO requirements that are applicable to them.~~

Insert 1

1

APPLICABLE SAFETY ANALYSES

monitor

The ~~SBFAS~~ is required to ~~filter~~ the ~~air space between the containment and shield building during a loss of coolant accident (LOCA), as discussed in FSAR, Chapter 15 (Ref. 1)~~.

Insert 2

1

fuel pool area radiation instrumentation

The ~~SBFAS~~ satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

1

LCO

The LCO on equipment OPERABILITY ensures that the ~~SBFAS~~ will perform as required when called upon.

fuel pool area radiation instrumentation

1

fuel pool area high radiation instrument logic trains

The LCO requires two ~~channels of SBFAS automatic and Manual Trip~~ to be OPERABLE. ~~Two channels are necessary to ensure the required redundancy should one channel become inoperable~~.

Insert 3

1

APPLICABILITY

~~The SBFAS must be OPERABLE in MODES 1, 2, 3, and 4, since the possibility of a LOCA is greatest in these MODES. In MODE 5 or 6 the probability of a LOCA is greatly diminished, and there is ample time to respond manually to a LOCA event.~~

Insert 4

1

ACTIONS

~~When the number of inoperable channels in a trip Function exceeds those specified in the Conditions associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.~~

1

1

INSERT 1

fuel pool area radiation instrumentation which actuates the Shield Building Ventilation System (SBVS) and aligns the system to the area of the spent fuel storage pool to filter airborne radioactive particulate following a fuel handling accident (FHA). The SBVS requirements are specified in LCO 3.6.9, "Shield Building Ventilation System (SBVS)." On a high fuel pool area radiation signal, the SBVS starts and aligns to exhaust the spent fuel storage pool area through the SBVS filtration units. In this alignment, the system achieves and maintains a negative air pressure in the fuel handling building. Six channels of fuel pool area high radiation instrumentation, with a sensing range of 1 to 10E4 mR/hr, are provided to the fuel pool area instrument logic. The fuel pool area instrument logic includes two independent, redundant actuation logic trains arranged in a two-out-of-three coincidence with input from the fuel pool area high radiation channels; three channels per train. Each SBVS train is actuated by its associated fuel pool area instrument logic train. The instrumentation requirements associated with other SBVS actuations are included in LCO 3.3.3, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," and LCO 3.3.4, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip." | R1

1

INSERT 2

spent fuel storage pool area during an FHA involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 72 hours). The analysis of the FHA is given in Reference 1

The FHA analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. Additionally, filtration of the fuel handling building is not assumed for an FHA in the spent fuel storage pool. The Technical Requirements Manual includes a decay time requirement that no fuel movement will commence until 72 hours after shutdown. This ensures that the FHA assumptions, including no filtration of the fuel handling building, are preserved. In the event an FHA in the spent fuel storage pool occurs within 72 hours of a unit shutdown, automatic actuation of the SBVS on fuel pool area high radiation will provide filtration to minimize the consequences of this improbable event.

1

INSERT 3

For the purpose of this LCO, each instrument logic train consists of at least two fuel pool area high radiation channels, associated actuation logic and relays necessary to actuate the associated SBVS train and valves.

1

INSERT 4

During movement of recently irradiated fuel assemblies in the spent fuel storage pool, the fuel pool area ventilation high radiation instrumentation is required to be OPERABLE to support the fuel pool area ventilation mode of the SBVS. Due to radioactive decay associated with irradiated fuel assemblies in the spent fuel storage pool, the SBVS is only required to mitigate FHAs that involve handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 72 hours).

When movement of irradiated fuel assemblies within the spent fuel storage pool is not being conducted, the potential for an FHA within the fuel pool area does not exist. Additionally, due to radioactive decay, an FHA that involves handling non-recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core longer than 72 hours) will result in doses that are well within the guideline values specified in 10 CFR 50.67 even without fuel pool area ventilation filtration capability. Therefore, under these conditions the fuel pool area ventilation mode of the SBVS, including the associated high radiation instrumentation, is not required to be OPERABLE.

BASES

ACTIONS (continued)

A.1

Insert 5

Condition A applies to the failure of one SBFAS Manual Trip channel or of one Actuation Logic associated with the Chemical and Volume Control System Isolation Signal or SBFAS. Required Action A.1 requires restoration of the inoperable channel to restore redundancy of the affected Function. The Completion Time of 48 hours is consistent with the Completion Time of other ESFAS Functions employing similar logic and should be adequate for most repairs while minimizing the risk of operating with an inoperable channel for a manually actuated Function. 1

B.1 and B.2

---------- REVIEWER'S NOTE ----------
Adoption of a MODE 4 end state requires the licensee to make the following commitments:

1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F]. 2

2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010. 2

Condition B specifies the shutdown track to be followed if the Required Action and associated Completion Time of Condition A are not met. If Required Action A.1 cannot be met within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. Inoperability of the SBFAS has no effect on plant core damage frequency in any operational MODE. 1

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 1

1

INSERT 5

When one fuel pool area high radiation instrument train is inoperable (e.g., one or two required fuel pool area high radiation channels are inoperable or an actuation logic relay is inoperable), the inoperable instrument train must be restored to OPERABLE status within 7 days. The remaining OPERABLE instrument train in this degraded condition is capable of actuating one SBVS train following an FHA. The Completion Time is consistent with the Completion Time to restore one inoperable SBVS train to OPERABLE status in the ACTIONS of Specification 3.6.9 and is based on consideration of such factors as the availability of the OPERABLE redundant instrument train and the low probability of an FHA occurring during this period.

B.1

When Required Action A.1 is not met within the required Completion Time or when two fuel pool area high radiation instrument trains are inoperable during movement of recently irradiated fuel assemblies, action must be taken to place the unit in a condition in which the LCO does not apply. This involves immediately suspending movement of recently irradiated fuel assemblies in the spent fuel storage pool. This does not preclude the movement of fuel to a safe position.

BASES

ACTIONS (continued)

~~Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~

1

~~The Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.~~

1

SURVEILLANCE REQUIREMENTS

SR 3.3.~~10~~.1

8

3

Insert 6

~~The SBFAS can be initiated either on a Safety Injection Actuation Signal (SIAS) or manually. This Surveillance is a restatement of SR 3.3.5.1 on the SIAS Function. Performing SR 3.3.5.1 satisfies this Surveillance. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. [The Frequency is the same as that for SR 3.3.5.1.~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

.

1 **INSERT 6**

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.~~10~~.2

8

fuel pool area high radiation instrumentation

3
1

A CHANNEL FUNCTIONAL TEST is performed on the ~~manual SBFAS~~ actuation circuitry. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. ~~This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions~~.

1

Insert 7

~~This Surveillance verifies that the trip push buttons are capable of opening contacts in the Actuation Logic as designed, de-energizing the initiation relays and providing Manual Trip of the Function. [The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at a Frequency of once every [18] months~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

2

Insert 8

~~----------~~]

1

REFERENCES

1. FSAR, Chapter ~~[~~15~~]~~.

U

1

2. ~~CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.~~

1

UFSAR, Section 7.3

1 **INSERT 7**

The CHANNEL FUNCTIONAL TEST tests the individual sensor subsystems using an analog test input to each bistable. A test signal is superimposed on the input in one channel at a time to verify that the bistable trips within the specified tolerance around the setpoint. Any setpoint adjustment shall be consistent with the assumptions of the plant specific setpoint analysis.

1 **INSERT 8**

SR 3.3.8.3 CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the extension analysis. The requirements for this review are outlined in Reference 2.

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.8, BASES, FUEL POOL AREA RADIATION INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS Specification number and title. Numbering is changed from 3.3.10 to 3.3.8 and “(Analog)” is deleted from the title.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.8, FUEL POOL AREA RADIATION INSTRUMENTATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 9

3.3.9, Post Accident Monitoring (PAM) Instrumentation

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

INSTRUMENTATION

POST (PAM)

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

PAM for each Function .9-1

LCO 3.3.9 3.3.3.8 The ~~accident monitoring~~ instrumentation ~~channels shown~~ in Table 3.3~~-11~~ shall be OPERABLE.

Applicability **APPLICABILITY:** MODES 1, 2 and 3.

ACTION:

Add proposed ACTIONS Note

A02

ACTIONS a. Actions per Table 3.3~~-11~~.

.9-1

SURVEILLANCE REQUIREMENTS

Add proposed Surveillance Requirements Note

A03

SR 3.3.9.1
SR 3.3.9.2 4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations in accordance with the Surveillance Frequency Control Program.

.9-1

POST

TABLE 3.3-~~1~~1

ACCIDENT MONITORING INSTRUMENTATION

FUNCTION

REQUIRED

CONDITIONS REFERENCED FROM REQUIRED ACTION D.1

A04

A05

	~~INSTRUMENT~~	~~TOTAL NO.~~ OF CHANNELS	~~MINIMUM CHANNELS OPERABLE~~	~~ACTION~~	
10	~~1~~. Pressurizer Water Level	2	~~1~~	~~1, 6~~ E	
	~~2. Auxiliary Feedwater Flow Rate~~	~~1/pump~~	~~1/pump~~	~~7~~	
	~~3. RCS Subcooling Margin Monitor~~	~~2~~	~~1~~	~~1, 6~~	
	~~4. PORV Position Indicator Acoustic Flow Monitor~~	~~1/valve~~	~~1/valve~~	~~2~~	R01
	~~5. PORV Block Valve Position Indicator~~	~~1/valve~~	~~1/valve~~	~~2~~	
	~~6. Safety Valve Position Indicator~~	~~1/valve~~	~~1/valve~~	~~3~~	
13, 14, 15 and 16	~~7~~. ~~Incore thermocouples~~ (Core Exit Temperature)	~~4/core quadrant~~	~~2/core quadrant~~	~~1, 6~~ E	A06
	~~8. Containment Sump Water Level (Narrow Range)~~	~~1*~~	~~1*~~	~~4, 5~~	R01
6	~~9~~. Containment ~~Sump~~ Water Level (Wide Range)	2	~~1~~	~~4, 5~~ E	M01
5	~~10~~. Reactor Vessel Level ~~Monitoring System~~	2~~**~~	~~1**~~	~~4, 5~~ F	
7	~~11~~. Containment Pressure	2	~~1~~	~~1, 6~~ E	

Add proposed Functions 1, 2, 3, 4, 8, 11, 12, and 17

Add proposed footnotes a, b and c

M02 R01 LA01

~~* The non-safety grade containment sump water level instrument may be substituted.~~

~~** Definition of OPERABLE: A channel is composed of eight (8) sensors in a probe, of which four (4) sensors must be OPERABLE.~~

TABLE 3.3-11 (continued)

.9-1

ACTION STATEMENTS

One or more Functions with one Required channel inoperable

ACTION 1 - With the number of OPERABLE channels less than the Total No. of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION A

required

Add proposed ACTION B

L01

ACTION 2 - With position indication inoperable, restore the inoperable indicator to OPERABLE status or close the associated PORV block valve and remove power from its operator within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information once per shift to determine valve position.

R01

Required Action and associated Completion Time of Condition A

ACTION 4 - With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to the specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION A

required

30 days

L01

LA02

ACTION B

Initiate action in accordance with Specification 5.6.4 immediately

A07

See ITS 5.6

One or more Functions with two required channels inoperable

ACTION 5 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

ACTION C

one

7 days

L02

LA02

ACTION D
ACTION E
ACTION F

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

Be in MODE 3 in 6 hours and MODE 4 in 12 hours

M01

R1

Initiate action in accordance with Specification 5.6.4 immediately

A07

See ITS 5.6

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

LA02

One or more Functions with two required channels inoperable

ACTION 6 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION C

one

7 days

L02

ACTION D
ACTION E

ACTION 7 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 72 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

R01

~~DELETED~~

INSTRUMENTATION

POST ACCIDENT

3/4.3.3 MONITORING INSTRUMENTATION

(PAM)

~~RADIATION MONITORING~~

LIMITING CONDITION FOR OPERATION

LCO 3.3.9

3.3.3.1 The ~~radiation monitoring~~ PAM instrumentation ~~channels shown~~ for each Function in Table 3.3-~~6~~ .9-1 shall be OPERABLE ~~with their alarm/trip setpoints within the specified limits~~. L03

APPLICABILITY: ~~As shown in Table 3.3-6.~~ MODES 1, 2, and 3. L04

ACTION:

~~a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.~~ L03

ACTIONS

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-~~6~~ .9-1.

~~c. The provisions of Specification 3.0.3 are not applicable.~~ M03

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1
SR 3.3.9.2

4.3.3.1 Each ~~radiation~~ accident monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK~~,~~ and CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program.

See ITS 3.3.6

4.3.3.2 In accordance with the Surveillance Frequency Control Program, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

See ITS 3.3.7

TABLE 3.3-6

.9-1

POST ACCIDENT RADIATION MONITORING INSTRUMENTATION

FUNCTION

REQUIRED

CONDITIONS REFERENCED FROM REQUIRED ACTION D.1

A04

A05

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION	
1. AREA MONITORS						
a. Fuel Storage Pool Area	1	*	≤ 15 mR/hr	10^{-1} – 10^{4} mR/hr	13	See ITS 3.3.8
b. Containment (CIS)	3	****	≤ 90 mR/hr	1 – 10^{5} mR/hr	16	See ITS 3.3.6
9 c. Containment Area – Hi Range (Radiation (high range))	2 → 1 M04	1, 2, 3, & 4 L04	≤ 10 R/hr L03	1 – 10^{7} R/hr LA03	15 F	R1
d. Control Room Isolation	1 per intake	ALL MODES	≤ 320 cpm	10 - 10^{7} cpm	17	See ITS 3.3.7
2. PROCESS MONITORS						
a. Containment						
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 – 10^{6} cpm	14	See ITS 3.4.15
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 – 10^{6} cpm	14	

* With fuel in the storage pool or building. (See ITS 3.3.8)

**** During movement of recently irradiated fuel assemblies within containment. (See ITS 3.3.6)

~~TABLE 3.3-6 (Continued)~~

~~TABLE NOTATION~~

ACTION 12 - DELETED

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION C

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within ~~72 hours~~, or:

Add proposed ACTIONS A and B

M04

R1

7 days

L02

ACTION D
ACTION F

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s),and

2) Prepare and submit a Special Report to the Commission pursuant to Specification6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

Initiate action in accordance with Specification 5.6.4 immediately

A07

See ITS 5.6

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

INSTRUMENTATION

POST (PAM)

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

LCO 3.3.9 3.3.3.6 The ~~accident monitoring~~ (PAM) instrumentation ~~channels shown~~ for each Function in Table 3.3-~~10~~ .9-1 shall be OPERABLE.

Applicability **APPLICABILITY:** MODES 1, 2 and 3.

ACTION:

Add proposed ACTIONS Note — A02

ACTION A — One or more Functions with one Required channel inoperable

a.* ~~With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either~~ restore ~~the inoperable~~ channel to OPERABLE status within 30 ~~7~~ days~~, or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours~~. — required

Add proposed ACTION B — L01

ACTION C — One or more Functions with two required channels inoperable

b.* ~~With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either~~ restore ~~the inoperable~~ channel~~s~~ to OPERABLE status within ~~48 hours~~ 7 days or one be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours. — L02

ACTION D
ACTION E

ACTION A — Required Action and associated Completion Time of Condition A not met,

c.** ~~With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-10, either~~ restore ~~the inoperable~~ channel to OPERABLE status within 30 days ~~7 days if repairs are feasible without shutting down~~ or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status. — required — L01 — See ITS 5.6 — LA02 — A07

ACTION B — Initiate action in accordance with Specification 5.6.4 immediately

ACTION C — One or more Functions with two required channels inoperable

d.** ~~With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either~~ restore ~~the inoperable~~ channel~~(s)~~ to OPERABLE status within ~~48 hours if repairs are feasible without shutting down~~ or: — one — L02 — 7 days — LA02 — M01

ACTION D
ACTION E
ACTION F

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status, and

3. ~~Restore the Channel to OPERABLE status at the next scheduled refueling.~~

Be in MODE 3 in 6 hours and MODE 4 in 12 hours — R1

Initiate action in accordance with Specification 5.6.4 immediately

A07 — See ITS 5.6 — LA02

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* Action statements do not apply to Reactor Vessel Level Monitoring System, ~~Containment Sump Water Level (narrow range)~~ and Containment Sump Water Level (wide range) instruments.

** Action statements apply only to Reactor Vessel Level Monitoring System, ~~Containment Sump Water Level (narrow range)~~ and Containment Sump Water Level (wide range) instruments.

R01 — A01

INSTRUMENTATION

POST (PAM)

ACCIDENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

Add proposed Surveillance Requirements Note

A03

SR 3.3.9.1
SR 3.3.9.2

4.3.3.6 Each accident monitoring instrumentation channel will be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations in accordance with the Surveillance Frequency Control Program.

A01

.9-1

POST

TABLE 3.3-~~10~~

ACCIDENT MONITORING INSTRUMENTATION

FUNCTION

CONDITIONS REFERENCED FROM REQUIRED ACTION D.1

A04

A05

	INSTRUMENT	REQUIRED ~~NUMBER~~ OF CHANNELS	~~MINIMUM CHANNELS~~ OPERABLE	
7	~~1~~. Containment Pressure	2	~~1~~	
	2. Reactor Coolant ~~Outlet~~ Temperature ~~– T_{Hot} (Wide Range)~~ (System Hot Leg)	2	~~1~~	
	3. Reactor Coolant ~~Inlet~~ Temperature ~~– T_{Cold} (Wide Range)~~ (System Cold Leg)	2	~~1~~	
	4. ~~Reactor Coolant~~ Pressure ~~– Wide Range~~ (Pressurizer)	2	~~1~~	
10	~~5.~~ Pressurizer Water Level	2	~~1~~	
	~~6. Steam Generator Pressure~~	~~2/steam generator~~	~~1/steam generator~~	R01
	~~7. Steam Generator Water Level – Narrow Range~~	~~1/steam generator~~	~~1/steam generator~~	
11	8. Steam Generator Water Level – Wide Range	2 ~~1~~/steam generator~~*~~	~~1/steam generator*~~	
	~~9. Refueling Water Tank Water Level~~	~~2~~	~~1~~	
	~~10. Auxiliary Feedwater Flow Rate (Each pump)~~	~~1/pump*~~	~~1/pump*~~	
	~~11. Reactor Cooling System Subcooling Margin Monitor~~	~~2~~	~~1~~	R01
	~~12. PORV Position/Flow Indicator~~	~~2/valve***~~	~~1/valve**~~	
	~~13. PORV Block Valve Position Indicator~~	~~1/valve**~~	~~1/valve**~~	
	~~14. Safety Valve Position/Flow Indicator~~	~~1/valve***~~	~~1/valve***~~	
	~~15. Containment Sump Water Level (Narrow Range)~~	~~1****~~	~~1****~~	
6	~~16.~~ Containment Water Level (Wide Range)	2	~~1~~	
13, 14, 15 and 16	17. ~~Incore Thermocouples~~ (Core Exit Temperature)	~~4/core quadrant~~	~~2/core quadrant~~	A06
	~~18~~. Reactor Vessel Level ~~Monitoring System~~	2~~*****~~	~~1*****~~	L01
5	Add proposed Functions 1, 8, 12, and 17	Add proposed footnotes a, b and c		M02

~~* These corresponding instruments may be substituted for each other.~~

~~** Not required if the PORV block valve is shut and power is removed from the operator.~~

~~*** If not available, monitor the quench tank pressure, level and temperature, and each safety valve/PORV discharge piping temperature at least once every 12 hours.~~

~~**** The non-safety grade containment sump water level instrument may be substituted.~~

~~***** Definition of OPERABLE: A channel consists of eight (8) sensors in a probe of which four (4) sensors must be OPERABLE.~~

R01

LA01

~~DELETED~~

INSTRUMENTATION

POST ACCIDENT

3/4.3.3 MONITORING INSTRUMENTATION

(PAM)

~~RADIATION MONITORING INSTRUMENTATION~~

LIMITING CONDITION FOR OPERATION

LCO 3.3.9

3.3.3.1 The ~~radiation monitoring~~ PAM instrumentation ~~channels shown~~ for each Function in Table 3.3-~~6~~ .9-1 shall be OPERABLE ~~with their alarm/trip setpoints within the specified limits~~. L03

APPLICABILITY: ~~As shown in Table 3.3-6.~~ MODES 1, 2, and 3. L04

ACTION:

~~a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.~~ L03

ACTIONS

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-~~6~~ .9-1.

~~c. The provisions of Specification 3.0.3 are not applicable.~~ M03

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1
SR 3.3.9.2

4.3.3.1 Each ~~radiation~~ accident monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK~~,~~ and CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program. See ITS 3.3.6

4.3.3.2 In accordance with the Surveillance Frequency Control Program, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

See ITS 3.3.7

A01

TABLE 3.3-6

.9-1

POST ACCIDENT

RADIATION MONITORING INSTRUMENTATION

FUNCTION

REQUIRED

CONDITIONS REFERENCED FROM REQUIRED ACTION D.1

A04

A05

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION	
1. AREA MONITORS						
a. Fuel Storage Pool Area						
i. Criticality and Ventilation System Isolation Monitor	4	*	≤ 20 mR/hr	10^{-1} – 10^{4} mR/hr	22	See ITS 3.3.8
b. Containment Isolation	3	****	≤ 90 mR/hr	1 – 10^{7} mR/hr	25	See ITS 3.3.6
9 c. Containment Area – Hi Range (Radiation (high range))	2 1 M04	1, 2, 3 & 4 L04	Not Applicable	1 - 10^{7} R/hr LA03	27 F	R1
d. Control Room Isolation	1 per intake	ALL MODES	≤ 320 cpm	10^{-7}– 10^{-2} µCi/cc	26	See ITS 3.3.7
2. PROCESS MONITORS						
a. Containment						
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10^{-7} – 10^{-2} µCi/cc	23	See ITS 3.4.15
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10 – 10^{7} cpm	23	

* With fuel in the storage pool or building. See ITS 3.3.8

**** During movement of recently irradiated fuel assemblies within containment. See ITS 3.3.6

~~TABLE 3.3-6 (Continued)~~

~~ACTION STATEMENTS~~

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION 24 - DELETED

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within ~~72 hours~~, or:

Add proposed ACTIONS A and B

M04 R1

7 days

L02

ACTION C

ACTION D
ACTION F

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

Initiate action in accordance with Specification 5.6.4 immediately

A07

See ITS 5.6

DISCUSSION OF CHANGES
ITS 3.3.9, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.3.3.8 and Unit 2 CTS 3.3.3.6 Actions provide the compensatory actions to take when PAM instrumentation is inoperable. ITS 3.3.9 ACTIONS provide the compensatory actions for inoperable PAM instrumentation. The ITS 3.3.9 ACTIONS include a Note that allows separate Condition entry for each Function. This changes CTS by providing a specific allowance to enter the Action for each inoperable PAM instrumentation Function.

This change is acceptable because it clearly states the current requirement. The CTS considers each PAM instrumentation Function to be separate and independent from the others. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 Unit 1 CTS 3.3.3.8 and Unit 2 CTS 3.3.3.6 Surveillance Requirements consist of a single surveillance (4.3.3.8 and 4.3.3.6, respectively). CTS Surveillances 4.3.3.8 and 4.3.3.6 state the defined test requirements (Channel Check and Channel Calibration) in accordance with the Surveillance Frequency Control Program. ITS 3.3.9 Surveillance Requirements consist of two separate surveillances SR 3.3.9.1 (Channel Check) and SR 3.3.9.2 (Channel Calibration). A note precedes the ITS surveillances and specifies the applicability of the surveillances to each PAM Function. In addition, ITS SR 3.3.9.2 contains a Note that excludes neutron detectors from the Channel Calibration requirement. Each ITS surveillance is applicable to the PAM Functions (including the newly added Functions) specified in ITS Table 3.3.9-1. This changes CTS by presenting the surveillance requirements in two distinct surveillance requirements and including an exception to the channel calibration requirement for neutron detectors. This change is acceptable as it results solely from the change in the format and presentation of the CTS surveillance requirements necessary to conform to the ITS. Separate change discussions address additional changes where applicable (e.g., addition of functions).

A04 Unit 1 CTS Table 3.3-11 and Unit 1 and 2 Tables 3.3-6 contain a column that specifies the applicable Action(s) for each instrument listed on the Table. The corresponding column in ITS Table 3.3.9-1 specifies the applicable Action Condition referenced from Required Action D.1. Unit 2 CTS 3.3.3.6 addresses Actions within the ACTION section and does not specify Actions for each Instrument (Function) individually in Unit 2 CTS Table 3.3-10. This changes the Unit 1 CTS by replacing the CTS table column specifying the applicable Actions for each PAM instrument with the ITS Table column specifying only the Action referenced from Required Action D.1 for each PAM Function. This changes the

Unit 2 CTS by replacing a table column and adding a table column specifying only the Action referenced from Required Action D.1 for each PAM instrument (Function).

In the ITS, ACTIONS except ACTIONS E and F are common to the PAM Functions listed on Table 3.3.9-1. Therefore, in the ITS, only ACTIONS E and F must be referenced from the Table as applicable for each PAM Function.

The proposed change to the Actions column in the PAM Table is acceptable because it is necessary to conform to the format and presentation of the PAM requirements in the ITS. Technical changes to the CTS Actions are identified and discussed in separate DOCs. The revision of the PAM table Action column does not introduce any additional technical changes and represents a necessary reformat to conform to the ITS. As the proposed change is the result of changes in the format and presentation of the PAM requirements, it is designated administrative.

A05 The title of Unit 1 CTS 3.3.3.8 and Unit 2 CTS 3.3.3.6, “Accident Monitoring Instrumentation” and CTS 3.3.3.1, “Radiation Monitoring Instrumentation” is revised to “Post Accident Monitoring (PAM) Instrumentation.” The CTS LCO operability requirement and Actions are revised from referencing instrument “channels” to referencing instrument “Functions.” The requirements for each PAM instrument (Function) listed in CTS Table 3.3-11 (Unit 1), CTS Table 3.3-10 (Unit 2) and CTS Table 3.3-6 are further revised to specify the "Required Channels" for each instrument Function. In addition, the CTS reference to "Minimum Channels Operable" on Table 3.3-11 (Unit 1), Table 3.3-10 (Unit 2) and CTS 3.3-6 is deleted. The proposed changes are consistent with the presentation of this information in the corresponding ITS 3.3.9, “Post Accident Monitoring (PAM) Instrumentation.” R1

The proposed changes are acceptable because they result in a uniform presentation of the PAM requirements without introducing a technical change to the requirements. The use of the term "Function" in the LCO and on ITS Tables 3.3.9-1 merely replaces the term "channel" in the LCO and "Instrument" in CTS Table 3.3-11 (Unit 1), CTS Table 3.3-10 (Unit 2) and CTS Table 3.3-6 to provide a uniform reference in the ITS. The corresponding proposed ITS Actions reference one or two inoperable channels associated with a PAM Function. As such, the proposed Actions are consistent with the use of the term Function and "Required Channels" used in the ITS to specify the instrument requirements. In addition, the proposed Actions, by addressing two or more inoperable channels (i.e., ITS Action C) effectively replace the CTS need to reference less than the "Minimum Operable Channels" in the Actions. Thus, the CTS "Minimum Operable Channels" column on Table 3.3-11, Table 3.3-10 and Table 3.3-6 is not needed and is deleted. The proposed changes are designated administrative because they represent a change in the format and presentation of the requirements and do not result in a technical change. R1 R1

A06 The CTS PAM requirement for Core Exit Temperature consists of specifying 4 incore thermocouples per core quadrant. The ITS presents this requirement differently. The corresponding ITS requirement for Core Exit Temperature specifies (2) Required Channels for each core quadrant and lists each quadrant R1

separately. In addition, the ITS requirement for two channels is modified by footnote (c) that requires each channel to contain two or more core exit thermocouples (CETs). The CTS is revised to conform to the ITS. This changes the CTS by revising the presentation of the Core Exit Temperature Function to be stated in terms of "Required Channels" per quadrant. | R1 | R1

The purpose of the PAM TS requirement is to ensure sufficient indication instrumentation is maintained operable to provide the necessary indication in post-accident conditions. The CTS requires (4) CETs per core quadrant. The ITS, by specifying (2) channels per quadrant with each channel consisting of (2) CETs, effectively requires the same number of CETs operable as the CTS. As such, the proposed change is acceptable because it continues to require an adequate number of operable CETs to assure the Core Exit Temperature indication continues to be available for PAM purposes. The proposed change only revises the presentation of this information to be consistent with the ITS. This change in format does not introduce a technical change to the CTS requirements. The proposed change is designated administrative because it does not introduce a technical change to the CTS.

A07 Unit 1 CTS Table 3.3-11 Actions 4 and 5, Unit 2 CTS 3.3.3.6 Actions c and d, and CTS Table 3.3-6 Unit 1 Action 15 and Unit 2 Action 27 require, in part, to prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 and includes detail on what to include in the report and the timing to submit the report. ITS 3.3.9 ACTIONS B and F require a similar action by stating “Initiate action in accordance with Specification 5.6.4,” but does not include the detail of what to include in the report or the timing to submit the report. These details are provided in Specification 5.6.4, “Post Accident Monitoring Report.” This changes the CTS by moving report detail to another Specification in the ITS.

The purpose of the CTS action is to ensure appropriate level of information is provided to the NRC when submitting a special PAM report. Moving these administrative details to another Specification is acceptable equivalent information in retained in the ITS. Any changes related to the CTS actions moved to ITS Section 5.6 are discussed in the ITS 5.6 Discussion of Changes. This change is a presentation preference and is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS Table 3.3-11, Instrument 9 (Containment Sump Water Level (Wide Range)) specifies Action 5 when less than the Minimum Operable Channels is not met (i.e., both required channels are inoperable). Unit 2 CTS 3.3.3.6, Action d applies to Table 3.3-10, Instrument 16 (Containment Water Level (Wide Range) when the number of OPERABLE channels is less than the Minimum Channels Operable requirements (i.e., both required channels are inoperable). These actions require, in part, to prepare and submit a Special Report to the Commission within 30 days if the instrument cannot be restored to OPERABLE status within 48 hours. For the same Function (ITS Table 3.3.9-1, Function 6), ITS 3.3.9 ACTION E requires the unit to be placed in MODE 3 within 6 hours and | R1

MODE 4 within 12 hours if one channel cannot be restored to OPERABLE status within 7 days. This changes the CTS by requiring a plant shutdown instead of preparing and submitting a report to the NRC.

The purpose of the action requirement when at least one channel of Containment Water Level (wide range) cannot be restored to OPERABLE status within the required time is to ensure remedial actions are provided that are acceptable for continued operation or a plant shutdown is required. ISTS 3.3.11 (ITS 3.3.9), ACTION F, which requires initiation of action to prepare and submit a PAM report to the NRC, is applicable to PAM instrument Functions that have alternate means of monitoring developed and tested. This change is necessary because the Containment Water Level (wide range) PAM instrument does not currently have an established alternate means developed and tested to monitor containment level during worst case post-accident conditions. Therefore, if at least one Containment Water Level (wide range) channel cannot be restored to OPERABLE status within the required Completion Time when two required channels are inoperable, ACTION E will require placing the unit in MODE 3 in 6 hours and MODE 4 in 12 hours. This change is designated as more restrictive because a plant shutdown is required in ITS that is not required in CTS for a similar condition. | R1

M02 Unit 1 CTS Table 3.3-11 does not require OPERABLE indication channels for Excore Neutron Flux, Reactor Coolant System Hot Leg Temperature, Reactor Coolant System Cold Leg Temperature, Pressurizer Pressure, Penetration Flow Path Containment Isolation Valve Position, Condensate Storage Tank Level and Containment Hydrogen Concentration. These are added to the Unit 1 CTS and shown in ITS Table 3.3.9-1, Functions 1, 2, 3, 4, 8, 11,12, and 17. Unit 2 CTS Table 3.3-10 does not require OPERABLE indication channels for Excore Neutron Flux, Penetration Flow Path Containment Isolation Valve Position, Steam Generator Water Level (Wide Range), Condensate Storage Tank Level and Containment Hydrogen Concentration. These are added to the Unit 2 CTS and shown in ITS Table 3.3.9-1, Functions 1, 8, 12, and 17. Two channels are provided for Excore Neutron Flux (Function 1). Two channels per loop are provided for Reactor Coolant System Hot and Cold Leg Temperature (Functions 2 and 3, respectively). Two channels are provided for Pressurizer Pressure (Function 4). Two channels per penetration flow path are provided for Penetration Flow Path Containment Isolation Valve Position (Function 8). This requirement is modified by two footnotes, footnotes (a) and (b). Footnote (a) does not require position indication for isolation valves whose penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange or check valve with flow through the valve secured. Footnote (b) requires only one position indication channel per penetration flow path with one installed channel located in the control room. One channel per steam generator is provided for Steam Generator Water Level (Wide Range) (Function 11). One channel is provided for Condensate Storage Tank Level (Function 12). Two channels are provided for Containment Hydrogen Concentration (Function 17). ITS 3.3.9 ACTION A has been added to cover the Condition when one or more Functions have one required channel inoperable. ITS 3.3.9 Required Action A.1 allows 30 days to restore the required channel to OPERABLE status. If this Required Action and associated Completion Time of Condition A is not met, then ITS Required Action B.1 requires the immediate initiation of the actions specified

in Specification 5.6.4. ITS 3.3.9 ACTION C has been added to cover the Condition when one or more Functions have two required channels inoperable. ITS 3.3.9 Required Action C.1 requires restoration of one channel to OPERABLE status within 7 days. If this cannot be met, then ITS 3.3.9 Condition D must be entered, which will then require entry into Condition E where ITS 3.3.9 Required Actions E.1 and E.2 require the unit to be in MODE 3 within 6 hours and MODE 4 within 12 hours. A Note has been added to the ACTIONS to allow Separate Condition entry for each Function. In addition, separate Condition entry is allowed within a Function as follows: (a) for Function 8 on a penetration flow path basis and b) for Function 11 on a steam generator basis. In addition, SRs are added for each Function. These SRs are a CHANNEL CHECK for each required instrumentation channel that is normally energized (SR 3.3.9.1) and a CHANNEL CALIBRATION (SR 3.3.9.2). For the CHANNEL CALIBRATION of the Excore Neutron Flux Function channels, SR 3.3.9.2 is modified by a note that states "Neutron detectors are excluded from CHANNEL CALIBRATION." This changes the CTS by adding new Functions, Footnotes, a Note, applicable ACTIONS, and SRs. R1

This change is acceptable because a plant specific evaluation has concluded that these instrumentation channels are required to provide the primary, unambiguous information to the operator necessary to perform manual actions for which no automatic controls exist and that are required for safety systems to accomplish their safety functions for design basis accident (DBA) events. The change is designated as more restrictive because 8 new instrumentation functions are added to the Unit 1 Technical Specifications and 4 new instrumentation functions are added to the Unit 2 Technical Specifications.

M03 CTS 3.3.3.1 Action c states that the provisions of Specification 3.0.3 are not applicable. ITS 3.3.9 does not include an LCO 3.0.3 exception. This changes the CTS by eliminating the CTS 3.0.3 exception. CTS 3.0.3 requires the unit to be shutdown when the requirements of LCOs and associated Actions are not satisfied. In CTS or ITS, an exception to CTS 3.0.3 (ITS LCO 3.0.3) is not provided to the PAM instrumentation Specification. The containment radiation high range instrumentation is a PAM instrument and is being appropriately moved to the PAM Instrumentation Specification. Eliminating the CTS 3.0.3 exception ensures that the operators are provided guidance regarding actions to take in the event the required PAM instrumentation is inoperable and associated ACTIONS are not satisfied within the required time periods. This change is designated as more restrictive because an explicit exception provided in the CTS is eliminated.

M04 CTS Table 3.3-6 Instrument 1.c (Containment Area – Hi Range), in part, requires one channel to be OPERABLE. ITS Table 3.3.9-1 Function 9 (Containment Radiation (high range)) requires two channels to be OPERABLE. ITS 3.3.9 also includes actions when one of the two Containment Radiation (high range) channels is inoperable. ACTION A requires the inoperable channel to be restored to OPERABLE status within 30 days and ACTION B requires immediately initiating action in accordance with Specification 5.6.4 (i.e., prepare and submit a report to the NRC) if the channel cannot be restored to OPERABLE status within 30 days. This changes the CTS by changing the number of

OPERABLE containment radiation high range monitoring channels from one to two and provides ACTIONS when one of two channels is inoperable.

The availability of PAM instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. The containment radiation high range instruments are essential PSL identified instruments addressing the recommendations of Regulatory Guide 1.97, as required by Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements." This change is acceptable because the ITS reflects the requirements for diversity and redundancy stated in Regulatory Guide 1.97 and NRC Generic Letter 82-33. This provides the operator an unambiguous source of information for decisions needed following design basis events. The change is designated as a more restrictive because the number of required channels for the containment radiation high range Function is increased from one to two. Remedial actions are provided consistent with actions in the ISTS associated with the inoperability of a redundant channel of other PAM instruments.

RELOCATED SPECIFICATIONS

R01 Unit 1 CTS Table 3.3-11 and Unit 2 CTS Table 3.3-10 provide requirements for Post-Accident Monitoring Instrumentation channels. Each individual post accident monitoring parameter has a specific purpose, however, the general purpose for all accident monitoring instrumentation is to ensure sufficient information is available following an accident to allow an operator to verify the response of automatic safety systems, and to take preplanned manual actions to accomplish a safe shutdown of the plant.

The NRC position on application of the screening criteria to post-accident monitoring instrumentation is documented in a letter dated May 9, 1988 from T.E. Murley (NRC) to the Owners Groups. Regulatory Guide (RG) 1.97, Type A variables provide primary information; i.e., information that is essential for the direct accomplishment of the specified manual actions (including long term recovery actions) for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs or transients. Additionally, it could not be confirmed that RG 1.97, non-Type A Category 1 variables are not of prime importance in limiting risk. Therefore, the NRC position is that the post-accident monitoring instrumentation list should contain Type A instruments and non-Type A Category 1 instruments specified in the plant's safety evaluation report (SER) on RG 1.97. Accordingly, this position has been applied to the PSL Units 1 and 2 RG 1.97 instruments. Those instruments meeting these criteria are retained in the Technical Specifications. The instruments not meeting these criteria will be relocated from the Technical Specifications to the Technical Requirements Manual (TRM). | R1

A review of the PSL Units 1 and 2 UFSAR and the NRC Regulatory Guide 1.97 Safety Evaluation for PSL Units 1 and 2 shows that the following Unit 1 CTS Table 3.3-11 and Unit 2 CTS Table 3.3.10 Instruments do not meet Category 1 or Type A requirements.

Unit 1 CTS Table 3.3-11

Instrument 2	Auxiliary Feedwater Flow Rate
Instrument 3	RCS Subcooling Margin Monitor
Instrument 4	PORV Position Indicator – Acoustic Flow Monitor
Instrument 5	PORV Block Valve Position Indicator
Instrument 6	Safety Valve Position Indicator
Instrument 8	Containment Sump Level

Unit 2 CTS Table 3.3-10

Instrument 6	Steam Generator Pressure
Instrument 7	Steam Generator Water Level – Narrow Range
Instrument 9	Refueling Water Tank Water Level
Instrument 10	Auxiliary Feedwater Flow Rate
Instrument 11	RCS Subcooling Margin Monitor
Instrument 12	PORV Position/Flow Indicator
Instrument 13	PORV Block Valve Position Indicator
Instrument 14	Safety Valve Position/Flow Indicator
Instrument 15	Containment Sump Water Level (Narrow Range)

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. These instruments are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary.

2. The monitored parameters are not process variables, design features, or operating restrictions that are initial conditions of a DBA or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

3. These instruments are not structures, systems, or components that are part of the primary success path or which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. These post-accident monitoring instrument functions are not addressed in the PSL PRA and do not represent a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been satisfied for instruments which do not meet Regulatory Guide 1.97 Type A or non-Type A, Category 1 variable requirements, the requirements for these instruments may be relocated outside the Technical Specifications. The Technical Specification requirements for these instruments will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as a relocation because the requirements for these instruments do not meet the criteria in 10 CFR 50.36(c)(2)(ii) and have been relocated to the TRM.

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS Table 3.3-11, Instrument 10 and Unit 2 CTS Table 3.3-10, Instrument 18, Reactor Vessel Level Monitoring channel OPERABILITY requirements are modified by Unit 1 footnote ** and Unit 2 footnote ***** that states "Definition of OPERABLE: A channel is composed of eight (8) sensors in a probe, of which four (4) sensors must be OPERABLE." ITS Table 3.3.9-1 Function 5 requires two channels to be OPERABLE but the details of what constitutes an OPERABLE channel are moved to the Bases. This changes the CTS by moving the details of what constitutes an OPERABLE channel to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for two channels of Reactor Vessel Level Indication to be OPERABLE and to perform CHANNEL CHECKS and CHANNEL CALIBRATIONS of the channels and ITS retains the definition of OPERABLE – OPERABILITY in ITS Section 1.1. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5 of the ITS. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS 3.3.3.8 Actions 4 and 5 and Unit 2 CTS 3.3.3.6 Actions c and d require restoring the channel to OPERABLE status within 7 days and 48 hours, respectively, “if repairs are feasible without shutting down.” In addition, Unit 1 CTS Action 5 and Unit 2 CTS Action d include the requirement to “Restore the Channel to OPERABLE status at the next scheduled refueling.” ITS SR 3.3.9 ACTIONS do not contain these details. This changes the CTS by removing the unnecessary detail in meeting the Action requirement. (See DOC L02 for additional discussion.)

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the shutdown requirement as applicable. This change is designated as a less restrictive removal of detail change because operational details for meeting Technical Specifications requirements are being removed from the Technical Specification.

LA03 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-6 for radiation monitoring instrumentation has a column specifying the Measurement Range and a specific measurement range for the containment radiation instrumentation. ITS 3.3.6 does not retain the "MEASUREMENT RANGE" column or the specific measurement range of the R1

containment radiation monitor channels. This changes the CTS by moving the instrument measurement range to the Bases. R1

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the number of required channels and the appropriate Condition to enter if a required channel becomes inoperable. The upper end of the measurement range for the containment radiation monitoring instrumentation is changed from 10^7 to 10^8 R/hr to be consistent with the current radiation monitor instrument measurement range. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 (*Category 3 - Relaxation of Completion Time*) Unit 1 CTS 3.3.3.8 Actions 1 and 4 and Unit 2 CTS 3.3.3.6 Actions a and c address a single inoperable channel in one or more PAM Functions. Unit 1 CTS Action 4 and Unit 2 CTS Action a and c are revised to extend the allowed restoration time of 7 days to 30 days. In addition, the Unit 1 CTS Action 1 and Unit 2 Action a requirement to place the plant in HOT SHUTDOWN (Mode 4) if the affected channel is not restored to operable status is replaced with ITS ACTION B. ITS ACTION B requires that action be “initiated immediately in accordance with Specification 5.6.4.” Specification 5.6.4 requires a report to be submitted to the NRC outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. This changes the CTS by removing the shutdown requirement and allowing continued operation with one inoperable channel provided the reporting requirement is initiated immediately.

The proposed changes are acceptable due to the passive function of the PAM instrumentation (indication only) and the ability to adequately respond to an accident or event using redundant or alternate indications. In addition, the Completion Time extension is justified based on the low likelihood of an event occurring within the allowed Completion Time that would require the affected PAM Function to be Operable. If the channel cannot be restored to Operable status within 30 days, ITS ACTION B requires initiation of action to submit a report to the NRC describing the plan to restore the function to operable status and the alternate method of monitoring during post-accident conditions. The change to submit a report to the NRC in lieu of a plant shutdown is acceptable, because it takes into account the adequacy of indication(s) available to monitor the required parameter. In addition, the proposed ITS Action Condition B requires a report to be issued to advise the NRC of the circumstances and expected duration of the inoperability. The proposed changes are designated less

restrictive as they result in additional time being allowed to restore the inoperable PAM instrumentation to operable status.

L02 (*Category 3 - Relaxation of Completion Time*) Unit 1 CTS Table 3.3-11 Actions 5 and 6 and Unit 2 CTS 3.3.3.6 Actions b and d address the condition of instruments with less than the minimum channels operable and requires the inoperable channel(s) to be restored to operable status in 48 hours. Unit 1 CTS Table 3.3-6 Action 15 and Unit 2 CTS Table 3.3-6 Action 27 address, in part, the condition of containment area hi range instrument with less than the minimum channels operable and requires the inoperable channel(s) to be restored to OPERABLE status in 72 hours. The corresponding ITS 3.3.9 Conditions C, D, E and F address one or more Functions with two or more inoperable channels. Required Action C.1 allows 7 days to restore all but one of the inoperable channels (which effectively leaves the inoperability within ITS Action Condition A for a single inoperable channel). Failure to meet Required Action C.1 results in entry into ITS Action Condition D. Required Action D.1 requires entry into the Action Condition referenced on Table 3.3.9-1 for the affected Function.

The significant changes introduced by the new ITS Actions used in place of Unit 1 CTS Table 3.3-11 Actions 5 and 6, Unit 2 CTS 3.3.3.6 Actions b and d, and CTS Table 3.3-6 Unit 1 Action 15 and Unit 2 Action 27 include:

- The extension of the 48 hour restoration time to 7 days (by ITS Action Condition C), and
- The extension of the 72 hour CTS restoration time to 7 days (by ITS Action Condition C).

The proposed change extends the existing limit on operation from 48 hours and 72 hours to 7 days for all instrument Functions. The proposed change is consistent with the industry standard operating limits in the ISTS (i.e., NUREG-1432) for this equipment. The changes are acceptable because they take into consideration the relatively low likelihood of an event that would require the use of the affected PAM Function, the availability of alternate means to obtain the information, and the fact that the indication is passive and does not provide any automatic protective actions. In addition, the proposed change provides additional restoration time that reduces the potential for a unit transient (i.e., reduces risk) introduced by a plant shutdown and restart due solely to inoperable indicating instruments that provide no automatic protective features. These changes are designated less restrictive because additional time is allowed in the ITS to restore channels to OPERABLE status than was allowed in the CTS.

L03 (*Category 1 – Relaxation of LCO Requirements*) CTS 3.3.3.1 and CTS Table 3.3-6, Instrument 1.c provide alarm setpoint requirements for the Containment Area Hi Range instrument. CTS 3.3.3.1 Action a provides actions to take when the Containment Area Hi Range instrument alarm setpoint exceeds the specified value. ITS 3.3.9 does not include alarm setpoint for the Containment Radiation (high range) instrumentation. This changes the CTS by eliminating the alarm setpoint requirements for the Containment Area Hi Range instrument, including the associated actions and alarm setpoint value.

The purpose of the Containment Area Hi Range PAM instrumentation is to provide the control room operator with indication of radiation in the containment

to monitor for the potential of significant radiation releases and to provide indication for release assessment used by operators in determining the need to invoke site emergency plans. This change is acceptable because the alarm setpoint is not assumed to acuate in any design basis accident or transient nor required to mitigate any design basis accident or transient. Additionally, the alarm setpoint is not necessary for the Containment Area Hi Range PAM instrumentation to perform its specified function of providing indication to the control room operators. ITS 3.3.9 requires the Containment Radiation (high range) instrumentation to be OPERABLE. The ITS 3.3.9 requirement and the definition of OPERABLE-OPERABILITY are adequate to ensure that the Containment Radiation (high Range) PAM instrumentation remains capable of providing indication to the control room operators. Additionally, this change does not preclude the plant from maintaining an active alarm setpoint to notify operators of high radiation conditions within the containment. This change is consistent with the ISTS and is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L04 *(Category 2 – Relaxation of Applicability)* CTS Table 3.3-6 requires the Containment Area Hi Range instrument to be OPERABLE in MODES 1, 2, 3, and 4. ITS 3.3.9 requires the Containment Radiation (high range) PAM Function to be OPERABLE in MODES 1, 2, and 3. This changes the CTS by deleting the requirements for the Function in MODE 4.

The purpose of the Containment Area Hi Range PAM instrumentation is to provide the control room operator with indication of radiation in the containment to monitor for the potential of significant radiation releases and to provide indication for release assessment used by operators in determining the need to invoke site emergency plans. This change is acceptable because the requirements continue to ensure that the instruments are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. The Containment Radiation (high range) PAM instrumentation is required to be OPERABLE in MODES 1, 2, and 3. This is acceptable because in MODES 4, 5, and 6, accidents of the type that would require these instruments are less likely to occur because of reduced temperature and pressure in the RCS and secondary system. This change is consistent with the ISTS and is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~11~~ 9 Post Accident Monitoring (PAM) Instrumentation ~~(Analog)~~ 2

3.3.3.8
3.3.3.1

LCO 3.3.~~11~~ 9 The PAM instrumentation for each Function in Table 3.3.~~11~~ 9-1 shall be OPERABLE. 2

Applicability
DOC L04

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
DOC A02
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-11 Actions 1 & 4 DOC M04 A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
DOC L01 DOC M04 B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.~~5~~ 4	Immediately 5
Table 3.3-11 Actions 5 & 6 Table 3.3-6 Action 15 DOC L02 C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
Table 3.3-11 Actions 5 & 6 Table 3.3-6 Action 15 DOC L02 D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.~~11~~ 9-1 for the channel.	Immediately 2

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-11 Action 6 E. As required by Required Action D.1 and referenced in Table 3.3.~~11~~-1. 9	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours 2
Table 3.3-11 Action 5 Table 3.3-6 Action 15 F. ~~[~~ As required by Required Action D.1 and referenced in Table 3.3.~~11~~-1. 9	F.1 Initiate action in accordance with Specification 5.6.~~5~~. 4	Immediately~~]~~ 3 5 2

SURVEILLANCE REQUIREMENTS

----------NOTE----------

DOC A03 These SRs apply to each PAM instrumentation Function in Table 3.3.~~11~~-1. 9 2

SURVEILLANCE		FREQUENCY
4.3.3.8 4.3.3.1 SR 3.3.~~11~~.1 9	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 2 3

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
4.3.3.8 4.3.3.1 SR 3.3.~~11~~.2 9	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	2 ~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 3

CTS

Table 3.3-11

Table 3.3.~~11~~-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
	1. ~~[Logarithmic]~~ Neutron Flux (wide range)	2	E
DOC M02	2. Reactor Coolant System Hot Leg Temperature	2 per loop	E
	3. Reactor Coolant System Cold Leg Temperature	2 per loop	E
	4. ~~Reactor Coolant System~~ Pressurizer Pressure ~~(wide range)~~	2	E
Instrument 10	5. Reactor Vessel Water Level	2	~~[~~F~~]~~
Instrument 9 DOC M01	6. Containment ~~Sump~~ Water Level (wide range)	2	E
Instrument 11	7. Containment Pressure (wide range)	2	E
DOC M02	8. Penetration Flow Path Containment Isolation Valve Position	2 per penetration flow path[(a)(b)]	E
Table 3.3-6 Instrument 1.c DOC M04 & L03	9. Containment ~~Area~~ Radiation (high range)	2	~~[~~F~~]~~
Instrument 1	10. Pressurize r Water Level	2	E
DOC M02	11. Steam Generator Water Level (wide range)	2 per steam generator	E
	12. Condensate Storage Tank Level	2	E
Instrument 7	13. Core Exit Temperature - Quadrant ~~[~~1~~]~~	2[(c)]	E
	14. Core Exit Temperature - Quadrant ~~[~~2~~]~~	2[(c)]	E
	15. Core Exit Temperature - Quadrant ~~[~~3~~]~~	2[(c)]	E
	16. Core Exit Temperature - Quadrant ~~[~~4~~]~~	2[(c)]	E
DOC M02	~~17. Auxiliary Feedwater Flow~~ Insert 1	~~2~~	~~E~~

DOC M02

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel consists of two or more core exit thermocouples.

~~REVIEWER'S NOTE~~

~~Table 3.3.11-1 shall be amended for each unit as necessary to list:~~
~~1. All Regulatory Guide 1.97, Type A instruments and~~
~~2. All Regulatory Guide 1.97, Category I, non-Type A instruments specified in the unit's Regulatory Guide 1.97, Safety Evaluation Report.~~

1

Insert 1

DOC M01	17.	Containment Hydrogen Concentration	2	E

CTS

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~11~~ 9 Post Accident Monitoring (PAM) Instrumentation ~~(Analog)~~ 2

3.3.3.6
3.3.3.1

LCO 3.3.~~11~~ 9 The PAM instrumentation for each Function in Table 3.3.~~11~~ 9-1 shall be OPERABLE. 2

Applicability
DOC L04

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
DOC A02
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Actions a & c DOC M04 A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
DOC L01 DOC M04 B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.~~5~~ 4	Immediately 5
Actions b & d Table 3.3-6 Action 27 DOC L02 C. One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
Actions b & d Table 3.3-6 Action 27 DOC L02 D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.~~11~~ 9-1 for the channel.	Immediately 2

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b E. As required by Required Action D.1 and referenced in Table 3.3.~~11~~-1. 9	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours 2
Action d.2 Table 3.3-6 Action 27 F. ~~[~~ As required by Required Action D.1 and referenced in Table 3.3.~~11~~-1. 9	F.1 Initiate action in accordance with Specification 5.6.~~5~~. 4	Immediately~~]~~ 3 5 2

SURVEILLANCE REQUIREMENTS

----------NOTE----------
DOC A03 These SRs apply to each PAM instrumentation Function in Table 3.3.~~11~~-1. 9 2

SURVEILLANCE		FREQUENCY
4.3.3.6 4.3.3.1 SR 3.3.~~11~~.1 9	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	~~[31 days~~ 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 3

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.3.3.6 4.3.3.1	SR 3.3.~~11~~.2 9	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ----------		2
		Perform CHANNEL CALIBRATION.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3

CTS

Table 3.3.~~11~~-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

Table 3.3-10

		FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
DOC M02	1.	~~[Logarithmic]~~ Neutron Flux (wide range)	2	E
Instrument 2	2.	Reactor Coolant System Hot Leg Temperature	2 per loop	E
Instrument 3	3.	Reactor Coolant System Cold Leg Temperature	2 per loop	E
Instrument 4	4.	~~Reactor Coolant System~~ Pressurizer Pressure ~~(wide range)~~	2	E
Instrument 18	5.	Reactor Vessel Water Level	2	~~[~~F~~]~~
Instrument 16 DOC M01	6.	Containment ~~Sump~~ Water Level (wide range)	2	E
Instrument 1	7.	Containment Pressure (wide range)	2	E
DOC M02	8.	Penetration Flow Path Containment Isolation Valve Position	2 per penetration flow path[(a)(b)]	E
Table 3.3-6 Instrument 1.c DOC M04 & L03	9.	Containment ~~Area~~ Radiation (high range)	2	~~[~~F~~]~~
Instrument 5	10.	Pressurize r Water Level	2	E
Instrument 8	11.	Steam Generator Water Level (wide range)	2 per steam generator	E
DOC M02	12.	Condensate Storage Tank Level	2	E
Instrument 17	13.	Core Exit Temperature - Quadrant ~~[~~1~~]~~	2[(c)]	E
	14.	Core Exit Temperature - Quadrant ~~[~~2~~]~~	2[(c)]	E
	15.	Core Exit Temperature - Quadrant ~~[~~3~~]~~	2[(c)]	E
	16.	Core Exit Temperature - Quadrant ~~[~~4~~]~~	2[(c)]	E
DOC M02	~~17.~~	~~Auxiliary Feedwater Flow~~ Insert 1	~~2~~	~~E~~

DOC M02

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel consists of two or more core exit thermocouples.

~~REVIEWER'S NOTE~~

~~Table 3.3.11-1 shall be amended for each unit as necessary to list:~~
~~1. All Regulatory Guide 1.97, Type A instruments and~~
~~2. All Regulatory Guide 1.97, Category I, non-Type A instruments specified in the unit's Regulatory Guide 1.97, Safety Evaluation Report.~~

1

Insert 1

DOC M01	17.	Containment Hydrogen Concentration	2	E

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.9, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The heading for ISTS 3.3.11 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2) and Chemical and Volume Control System (CVCS) Isolation Signal (ISTS 3.3.9). Therefore, ISTS 3.3.2 and ISTS 3.3.9 are not included in the PSL ITS. The successive Specifications are renumbered as necessary.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Changes have been made to reflect change to numbering in ISTS Specification 5.6.5.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~11~~ 9 Post Accident Monitoring (PAM) Instrumentation ~~(Analog)~~ 2

BASES

BACKGROUND

The primary purpose of the post accident monitoring (PAM) instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety Functions for Design Basis Events.

The OPERABILITY of the PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident.

The availability of PAM instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by ~~plant specific documents (Ref. 1)~~ in Reference 1 addressing the recommendations of Regulatory Guide 1.97 (Ref. 2), as required by Supplement 1 to NUREG-0737, "TMI Action Items" (Ref. 3). 1

Type A variables are included in this LCO because they provide the primary information required to permit the control room operator to take specific manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs). ~~Because the list of Type A variables differs widely between plants, Table 3.3.11-1, in the accompanying LCO, contains no examples of Type A variables, except for those that may also be Category I.~~ 1

Category I variables are the key variables deemed risk significant because they are needed to:

- Determine whether other systems important to safety are performing their intended functions,
- Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release, and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

BASES

BACKGROUND (continued)

These key variables are identified by plant specific Regulatory Guide 1.97 analyses (Ref. 1). These analyses identified the plant specific Type A and Category I variables and provided justification for deviating from the NRC proposed list of Category I variables.

~~REVIEWER'S NOTE~~

~~Table 3.3.11-1, in the accompanying LCO, provides a list of variables typical of those identified by plant specific Regulatory Guide 1.97 analyses. Table 3.3.11-1 in the plant specific Technical Specifications shall list all Type A and Category I variables identified by plant specific Regulatory Guide 1.97 analyses, as amended by NRC's Safety Evaluation Report (SER) (Ref. 4). The specific instrument Functions listed in Table 3.3.11-1 are discussed in the LCO Bases.~~ 4

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the OPERABILITY of Regulatory Guide 1.97 Type A variables, so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of DBAs and
- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions.

The PAM instrumentation also ensures OPERABILITY of Category I, non-Type A variables. This ensures the control room operating staff can:

- Determine whether systems important to safety are performing their intended functions,
- Determine the potential for causing a gross breach of the barriers to radioactivity release,
- Determine if a gross breach of a barrier has occurred, and
- Initiate action necessary to protect the public as well as to obtain an estimate of the magnitude of any impending threat.

BASES

APPLICABLE SAFETY ANALYSES (continued)

PAM instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 meets Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Category I, non-Type A PAM instruments are retained in the Specification because they are intended to assist operators in minimizing the consequences of accidents. Therefore, these Category I variables are important in reducing public risk.

LCO

LCO 3.3.~~11~~ (9) requires two OPERABLE channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following that accident. (2)

Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

~~[More than two channels may be required at some units if the Regulatory Guide 1.97 analysis determined that failure of one PAM channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or to fail to accomplish a required safety function.]~~ (3)

The exception to the two channel requirement is Penetration Flow Path Containment Isolation Valve Position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of the passive valve or via system boundary status. If a normally active containment isolation valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE. (1)

The applicable CIVs are identified in UFSAR Table 6.2-16.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.~~11~~-1. ~~These discussions are intended as examples of what should be provided for each Function when the plant specific list is prepared.~~ (9) (2) (1)

BASES

LCO (continued)

1. ~~[Logarithmic]~~ Excore (wide range) Neutron Flux

~~[Logarithmic]~~ Neutron Flux ~~indication~~ is a Type A and Category 1 variable provided to verify reactor shutdown. 1 3

Excore

~~[At this unit, the [Logarithmic]~~ Neutron Flux PAM channels consist of ~~the following:]~~

Insert 1

2, 3. Reactor Coolant System (RCS) Hot and Cold Leg Temperature

Type A and

RCS Hot and Cold Leg Temperatures are Category I variables provided for verification of core cooling and long term surveillance.

coolant loop consists of two channels per reactor coolant loop.

Reactor outlet temperature ~~inputs to the PAM are provided by two fast response resistance elements~~ and associated transmitters ~~s~~ in each loop. The channels provide indication over a range of ~~32~~ 212 °F to 70~~0~~ °F.

Each channel consists of a resistance temperature detector

5 Insert 2

4. ~~Reactor Coolant System~~ Pressurizer Pressure ~~(wide range)~~ R1

Type A and

Pressurizer ~~RCS wide range~~ pressure is a Category I variable provided for verification of core cooling and RCS integrity long term surveillance.

Pressurizer

~~Wide range RCS loop~~ pressure is measured by pressure transmitters with a span of 0 ~~psig~~ to 3000 psi~~g~~. The pressure transmitters are located ~~o~~utside the containment. Redundant monitoring capability is provided by two trains of instrumentation. Control room indications are provided through the ~~inadequate core cooling (ICC) plasma~~ QSPDS display. The ~~ICC plasma~~ display is the primary indication used by the operator during an accident. Therefore, the PAM instrumentation LCO deals specifically with this portion of the instrument channel. R1 1

a

in

Pressurizer

~~In some plants, RCS~~ pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture or small break loss of coolant accident (LOCA). Operator actions to maintain a controlled cooldown, such as adjusting steam generator pressure or level, would use this indication. Furthermore, ~~RCS~~ pressure is one factor that may be used in decisions to terminate reactor coolant pump operation.

Pressurizer

1

Insert 1

two channels; each consisting of a fission chamber detector assembly providing continuous independent indication in the control room and on the hot shutdown panel. This signal is processed for a wide range of 2 x 10^{-8} to 200% power. | R1

1

Insert 2

Channel output is provided to the Qualified Safety Parameter Display System (QSPDS) which provides control room indication. Although the temperature range of 212°F to 705°F deviates from RG 1.97 Rev. 3 criteria. For temperatures where RCS temperature is below 350°F the shutdown cooling system would be in operation and hot leg temperatures would be closely represented by the core exit thermocouples which have a range of 32°F to 2300°F and cold leg temperature would be closely represented by the shutdown cooling system temperature element that has a range of 0°F to 400°F, therefore the range of 212°F to 705°F is acceptable (Ref. 4).

BASES

LCO (continued)

5. Reactor Vessel Water Level

Reactor Vessel Water Level is provided for verification and long term surveillance of core cooling. (a Category 1 variable) 1

The Reactor Vessel Water Level monitoring system provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory. The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it functions during the recovery interval. Therefore, it is designed to survive the high steam temperature that may occur during the preceding core recovery interval.

The level range extends from the top of the vessel down to the top of the fuel alignment plate. The response time is short enough to track the level during small break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot leg elevation, and the lowest levels just above the alignment plate. This provides the operator with adequate indication to track the progression of the accident and to detect the consequences of its mitigating actions or the functionality of automatic equipment.

Each channel consists of eight sensors in a probe. A channel is OPERABLE if four sensors, with one sensor in the upper head and three sensors in the upper plenum, are OPERABLE.

6. Containment ~~Sump~~ Water Level (wide range) 1

Containment ~~Sump~~ Water Level is provided for verification and long term surveillance of RCS integrity. (a Category 1 variable)

~~[~~ ~~For this unit,~~ Containment ~~Sump~~ Water Level instrumentation consists of ~~the following:~~ ~~]~~ 3
Insert 3

7. Containment Pressure (wide range)

Containment Pressure is provided for verification of RCS and containment OPERABILITY. (a Category 1 variable) 1

~~[~~ ~~For this unit,~~ Containment Pressure instrumentation consists of ~~the following:~~ ~~]~~ 3
Insert 4

1 3

Insert 3

two channels. Each channel consists of a level transmitter with output to an indicator in the control room. One channel is recorded. The wide range containment level transmitters are located inside containment. The wide range containment water level monitors have a range of -1 ft (bottom of the containment) to elevation 26 feet 1 inch of the containment.

Insert 4 1 3

two channels. Each channel consists of a pressure transmitter mounted outside containment with output to an indicator in the control room. One channel is recorded. The containment wide range pressure monitors have a range of -5 to 175 psig.

BASES

LCO (continued)

8. Penetration Flow Path Containment Isolation Valve Position

Penetration Flow Path Containment Isolation Valve (CIV) Position is provided for verification of containment OPERABILITY.

a Category 1 variable

CIV position is provided for verification of containment integrity. In the case of CIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active CIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the CIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE. Each penetration is treated separately and each penetration flow path is considered a separate function. Therefore, separate Condition entry is allowed for each inoperable penetration flow path.

~~[For this unit, the CIV position PAM instrumentation consists of the following:]~~

9. Containment ~~Area~~ Radiation (high range)

a Category 1 variable

Containment ~~Area~~ Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

~~[For this unit,~~ Containment ~~Area~~ Radiation instrumentation consists of ~~the following:]~~

Insert 5

Insert 5 1 3

two channels. Each channel consists of an ambient radiation detector mounted inside containment transmitting to an indicator in the control room. One channel is recorded. The containment high range radiation monitors have a range of 1 r/hr to 10^8 r/hr. |R1

BASES

LCO (continued)

10. Pressurizer Level

Water

Water

a Category 1 variable

R1

1

Pressurizer Level is used to determine whether to terminate safety injection (SI), if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the plant conditions necessary to establish natural circulation in the RCS and to verify that the plant is maintained in a safe shutdown condition.

Water

R1

~~[For this unit,~~ Pressurizer Level instrumentation consists of ~~the following:]~~

3

Insert 6

11. Steam Generator Water Level

a Type A and Category 1 variable

Steam Generator Water Level is provided to monitor operation of decay heat removal via the steam generators. The Category I indication of steam generator level ~~is the extended startup range level instrumentation. The extended startup range level~~ covers a span of ~~0~~ inches to ~~394~~ inches ~~above the lower tubesheet~~. The measured differential pressure is displayed ~~in inches of water at 68°F~~. ~~Temperature compensation of this indication is performed manually by the operator.~~ Redundant monitoring capability is provided by two trains of instrumentation. ~~The uncompensated level signal is input to the plant computer, a control room indicator, and the [Auxiliary Feedwater (AFW)] Control System~~.

1

0

132

as 0 – 100% level.

1

converted and

O

~~At some plants,~~ operator action is based on the control room indication of Steam Generator Water Level. The RCS response during a design basis small break LOCA is dependent on the break size. ~~For a certain range of break sizes, the boiler condenser mode of heat transfer is necessary to remove decay heat. At these plants,~~ Steam Generator Water L ~~extended startup range~~ level is a Type A variable because the operator must manually raise and control the steam generator level to establish ~~boiler condenser heat transfer~~. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated ~~extended startup range~~ level reaches the ~~boiler condenser setpoint~~.

1

steaming

minimum required level

1

Insert 6

3

two channels. Each channel consists of a differential pressure instrument transmitting to an indicator in the control room. One channel is recorded. The pressurizer water level instrument has a calibration range of 175 inches to 349 inches (0 – 100% indicated level). |R1

BASES

LCO (continued)

12. Condensate Storage Tank (CST) Level

a Category 1 variable

CST Level is provided to ensure water supply for ~~[~~AFW~~]~~. The CST provides the ensured safety grade water supply for the ~~[~~AFW~~]~~ System. ~~The CST consists of two identical tanks connected by a common outlet header.~~ Inventory is monitored by a 0 to ~~144 inch~~ 40 foot level indication ~~for each tank~~. CST Level is displayed on a control room indicator~~, strip chart recorder, and plant computer~~. In addition, a control room annunciator alarms on low level. 1 3 1

~~At some plants, CST Level is considered a Type A variable because the control room meter and annunciator are considered the primary indication used by the operator. The DBAs that require [AFW] are the loss of electric power, steam line break (SLB), and small break LOCA~~. The CST is the initial source of water for the ~~[~~AFW~~]~~ System. However, as the CST is depleted, manual operator action is necessary to replenish the CST or align suction to the ~~[AFW] pumps from the hotwell~~. 3 3 1

Unit 2 CST

13, 14, 15, 16. Core Exit Temperature

a Category 1 variable

Core Exit Temperature is provided for verification and long term surveillance of core cooling. 1

An evaluation was made of the minimum number of valid core exit thermocouples necessary for inadequate core cooling detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core uncovery and trend the ensuing core heatup. The evaluations account for core nonuniformities including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate or inadequate core cooling detection is ensured with two valid core exit thermocouples per quadrant.

The design of the Incore Instrumentation System includes a Type K (chromel alumel) thermocouple within each of the 56 incore instrument detector assemblies.

LCO (continued)

The junction of each thermocouple is located a few inches above the fuel assembly, inside a structure that supports and shields the incore instrument detector assembly string from flow forces in the outlet plenum region. These core exit thermocouples monitor the temperature of the reactor coolant as it exits the fuel assemblies.

The core exit thermocouples have a usable temperature range from 32°F to 2300°F, although accuracy is reduced at temperatures above 1800°F.

17. ~~[Auxiliary Feedwater (AFW)] Flow~~ Insert 7 3

~~[AFW] Flow is provided to monitor operation of decay heat removal via the steam generators.~~

~~The [AFW] Flow to each steam generator is determined from a differential pressure measurement calibrated to a span of 0 gpm to 1200 gpm. Redundant monitoring capability is provided by two independent trains of instrumentation for each steam generator. Each differential pressure transmitter provides an input to a control room indicator and the plant computer. Since the primary indication used by the operator during an accident is the control room indicator, the PAM instrumentation Specification deals specifically with this portion of the instrument channel.~~

~~At some plants [AFW] Flow is a Type A variable because operator action is required to throttle flow during an SLB accident in order to prevent the [AFW] pumps from operating in runout conditions. [AFW] Flow is also used by the operator to verify that the [AFW] System is delivering the correct flow to each steam generator. However, the primary indication used by the operator to ensure an adequate inventory is steam generator level.~~ 1

Two channels are required to be OPERABLE for all but one Function. Two OPERABLE channels ensure that no single failure, within either the PAM instrumentation or its auxiliary supporting features or power sources (concurrent with the failures that are a condition of or result from a specific accident), prevents the operators from being presented the information necessary for them to determine the safety status of the plant and to bring the plant to and maintain it in a safe condition following that accident.

1

Insert 7

Containment Hydrogen Concentration

Containment Hydrogen Concentration is a Type A and Category 1 variable provided to allow an ability to measure containment hydrogen concentration without laboratory analysis. The channels provide indication over a range of 0 – 10%.

BASES

LCO (continued)

In Table 3.3.~~11~~-1 the exception to the two channel requirement is Containment Isolation Valve Position.

Two OPERABLE channels of core exit thermocouples are required for each channel in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number ~~and locations~~ provided for diagnosis of local core problems. ~~Therefore, two randomly selected thermocouples may not be sufficient to meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one be located near the center of the core and the other near the core perimeter, such that the pair of core exit thermocouples indicate the radial temperature gradient across their core quadrant. Plant specific evaluations in response to Item II.F.2 of NUREG-0737 should have identified the thermocouple pairings that satisfy these requirements.~~ Two sets of two thermocouples in each quadrant ensure a single failure will not disable the ability to determine the radial temperature gradient.

Core uniformities including in-core effects of the radial decay power distribution, ex-core effects of condensate runback in the hot legs and non-uniform inlet temperatures were accounted for in the evaluation.

For loop and steam generator related variables, the required information is individual loop temperature and individual steam generator level. In these cases two channels are required to be OPERABLE for each loop of steam generator to redundantly provide the necessary information.

In the case of Containment Isolation Valve Position, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of the passive valve or via system boundary status. If a normally active containment isolation valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, plant conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

BASES

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.~~11~~-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function. When the Required Channels in Table 3.3.~~11~~-1 are specified (e.g., on a ~~per steam line,~~ per loop, per SG, etc., basis) then Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

This Required Action specifies initiation of actions in accordance with Specification 5.6.~~5~~, which requires a written report to be submitted to the Nuclear Regulatory Commission. This report discusses the cause of the inoperability and identifies proposed restorative Required Actions. This Required Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Required Actions are identified before a loss of functional capability condition occurs.

C.1

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable

BASES

ACTIONS (continued)

because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

D.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.~~11~~-1. The applicable Condition referenced in the Table is Function dependent. Each time Required Action C.1 is not met, and the associated Completion Time has expired, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition. 9 2

E.1 and E.2

If the Required Action and associated Completion Time of Condition D is not met, and Table 3.3.~~11~~-1 directs entry into Condition E, the plant must be brought to a MODE in which the requirements of this LCO do not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. 9 2

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

~~[At this plant,~~ alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the plant, but rather to follow the directions of Specification 5.6.~~5~~. 4 3 5 The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.~~]~~

BASES

SURVEILLANCE REQUIREMENTS

A Note at the beginning of the Surveillance Requirements specifies that the following SRs apply to each PAM instrumentation Function in Table 3.3.~~11~~-1. 9 2

SR 3.3.~~11~~.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

~~[The Frequency of 31 days is based upon plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is a rare event. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel during normal operational use of the displays associated with this LCO's required channels.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.3.~~11~~.2

A CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies the channel responds to the measured parameter within the necessary range and accuracy. A Note allows exclusion of neutron detectors from the CHANNEL CALIBRATION.

[~~At this unit,~~ CHANNEL CALIBRATION shall find measurement errors are within the following acceptance criteria: ~~]~~

For the Containment ~~Area~~ Radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr, and a one point calibration check of the detector below 10 R/hr with a gamma source. | R1

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD) sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

[~~The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by an [18] month calibration interval for the determination of the magnitude of equipment drift.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 3

REFERENCES

1. ~~Plant specific document (e.g., FSAR, NRC Regulatory Guide 1.97, SER letter).~~ St. Lucie Unit No. 1 Engineering Evaluation of Instrumentation System for Regulatory Guide 1.97, Rev. 3 (ML17216A336).
2. Regulatory Guide 1.97, , Rev. 3
3. NUREG-0737, Supplement 1.
4. NRC Safety Evaluation Report (SER).

1

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~11~~ 9 Post Accident Monitoring (PAM) Instrumentation ~~(Analog)~~ 2

BASES

BACKGROUND

The primary purpose of the post accident monitoring (PAM) instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety Functions for Design Basis Events.

The OPERABILITY of the PAM instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident.

The availability of PAM instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by ~~plant specific documents (Ref. 1)~~ in Reference 1 addressing the recommendations of Regulatory Guide 1.97 (Ref. 2), as required by Supplement 1 to NUREG-0737, "TMI Action Items" (Ref. 3). 1

Type A variables are included in this LCO because they provide the primary information required to permit the control room operator to take specific manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs). ~~Because the list of Type A variables differs widely between plants, Table 3.3.11-1, in the accompanying LCO, contains no examples of Type A variables, except for those that may also be Category I.~~ 1

Category I variables are the key variables deemed risk significant because they are needed to:

- Determine whether other systems important to safety are performing their intended functions,
- Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release, and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

BASES

BACKGROUND (continued)

These key variables are identified by plant specific Regulatory Guide 1.97 analyses (Ref. 1). These analyses identified the plant specific Type A and Category I variables and provided justification for deviating from the NRC proposed list of Category I variables.

~~REVIEWER'S NOTE~~

~~Table 3.3.11-1, in the accompanying LCO, provides a list of variables typical of those identified by plant specific Regulatory Guide 1.97 analyses. Table 3.3.11-1 in the plant specific Technical Specifications shall list all Type A and Category I variables identified by plant specific Regulatory Guide 1.97 analyses, as amended by NRC's Safety Evaluation Report (SER) (Ref. 4). The specific instrument Functions listed in Table 3.3.11-1 are discussed in the LCO Bases.~~ 4

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the OPERABILITY of Regulatory Guide 1.97 Type A variables, so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of DBAs and
- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions.

The PAM instrumentation also ensures OPERABILITY of Category I, non-Type A variables. This ensures the control room operating staff can:

- Determine whether systems important to safety are performing their intended functions,
- Determine the potential for causing a gross breach of the barriers to radioactivity release,
- Determine if a gross breach of a barrier has occurred, and
- Initiate action necessary to protect the public as well as to obtain an estimate of the magnitude of any impending threat.

BASES

APPLICABLE SAFETY ANALYSES (continued)

PAM instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 meets Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Category I, non-Type A PAM instruments are retained in the Specification because they are intended to assist operators in minimizing the consequences of accidents. Therefore, these Category I variables are important in reducing public risk.

LCO

LCO 3.3.~~11~~ (9) requires two OPERABLE channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following that accident. (2)

Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

~~[More than two channels may be required at some units if the Regulatory Guide 1.97 analysis determined that failure of one PAM channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or to fail to accomplish a required safety function.]~~ (3)

The exception to the two channel requirement is Penetration Flow Path Containment Isolation Valve Position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of the passive valve or via system boundary status. If a normally active containment isolation valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE. (1)

The applicable CIVs are identified in UFSAR Table 6.2-52.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.~~11~~-1. ~~These discussions are intended as examples of what should be provided for each Function when the plant specific list is prepared.~~ (9) (2) (1)

BASES

LCO (continued)

1. ~~[Logarithmic]~~ Excore (wide range) Neutron Flux

~~[Logarithmic]~~ Neutron Flux ~~indication~~ is a Category 1 variable provided to verify reactor shutdown.

Excore

~~[At this unit, the [Logarithmic]~~ Neutron Flux PAM channels consist of ~~the following:]~~

Insert 1

1
3

2, 3. Reactor Coolant System (RCS) Hot and Cold Leg Temperature

Type A and

RCS Hot and Cold Leg Temperatures are Category I variables provided for verification of core cooling and long term surveillance.

coolant loop

consists of two channels per reactor coolant loop.

Each channel consists of a resistance temperature detector

Reactor outlet temperature ~~inputs to the PAM are provided by two fast response resistance elements~~ and associated transmitter~~s~~ in each loop. The channels provide indication over a range of ~~32~~ °F to 7~~0~~00°F.

50

5

Insert 2

4. ~~Reactor Coolant System~~ Pressurizer Pressure ~~(wide range)~~

Pressurizer

Type A and

~~RCS~~ ~~wide range~~ pressure is a Category I variable provided for verification of core cooling and RCS integrity long term surveillance.

Pressurizer

~~Wide range RCS loop~~ pressure is measured by pressure transmitters with a span of 0 ~~psig~~ to 3000 psi~~g~~. The pressure transmitters are located ~~o~~utside the containment. Redundant monitoring capability is provided by two trains of instrumentation. Control room indications are provided through the ~~inadequate core cooling (ICC) plasma~~ display. The ~~ICC plasma~~ display is the primary indication used by the operator during an accident. Therefore, the PAM instrumentation LCO deals specifically with this portion of the instrument channel.

a

in

QSPDS

Pressurizer

~~In some plants, RCS~~ pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture or small break loss of coolant accident (LOCA). Operator actions to maintain a controlled cooldown, such as adjusting steam generator pressure or level, would use this indication. Furthermore, ~~RCS~~ pressure is one factor that may be used in decisions to terminate reactor coolant pump operation.

Pressurizer

R1

1

1

Insert 1

two channels; each consisting of a fission chamber detector assembly providing continuous independent indication in the control room and on the hot shutdown panel. This signal is processed for a wide range of 2 x 10^{-8} to 200% power. |R1

1

Insert 2

Channel output is provided to the Qualified Safety Parameter Display System (QSPDS) which provides control room indication.

BASES

LCO (continued)

5. Reactor Vessel Water Level

Reactor Vessel Water Level is provided for verification and long term surveillance of core cooling. (a Category 1 variable) 1

The Reactor Vessel Water Level monitoring system provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory. The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it functions during the recovery interval. Therefore, it is designed to survive the high steam temperature that may occur during the preceding core recovery interval.

The level range extends from the top of the vessel down to the top of the fuel alignment plate. The response time is short enough to track the level during small break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot leg elevation, and the lowest levels just above the alignment plate. This provides the operator with adequate indication to track the progression of the accident and to detect the consequences of its mitigating actions or the functionality of automatic equipment.

Each channel consists of eight sensors in a probe. A channel is OPERABLE if four sensors, with one sensor in the upper head and three sensors in the upper plenum, are OPERABLE.

6. Containment ~~Sump~~ Water Level (wide range) 1

Containment ~~Sump~~ Water Level is provided for verification and long term surveillance of RCS integrity. (a Category 1 variable)

~~[~~ ~~For this unit,~~ Containment ~~Sump~~ Water Level instrumentation consists of ~~the following:~~ ~~]~~ 3
Insert 3

7. Containment Pressure (wide range)

Containment Pressure is provided for verification of RCS and containment OPERABILITY. (a Category 1 variable) 1

~~[~~ ~~For this unit,~~ Containment Pressure instrumentation consists of ~~the following:~~ ~~]~~ 3
Insert 4

Insert 3 (1) (3)

two channels. Each channel consists of a level transmitter with output to an indicator in the control room. Both channels are recorded. The wide range containment level transmitters are located outside containment. The wide range containment water level monitors have a range of -1 foot (bottom of the containment) to elevation 26 feet of the containment.

Insert 4 (1) (3)

two channels. Each channel consists of a pressure transmitter mounted outside containment with output to an indicator in the control room. One channel is recorded. The containment wide range pressure monitors have a range of -5 to 175 psig.

BASES

LCO (continued)

8. Penetration Flow Path Containment Isolation Valve Position

a Category 1 variable

Penetration Flow Path Containment Isolation Valve (CIV) Position is provided for verification of containment OPERABILITY. 1

CIV position is provided for verification of containment integrity. In the case of CIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active CIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the CIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE. Each penetration is treated separately and each penetration flow path is considered a separate function. Therefore, separate Condition entry is allowed for each inoperable penetration flow path.

~~[For this unit, the CIV position PAM instrumentation consists of the following:]~~ 3

9. Containment ~~Area~~ Radiation (high range) R1

a Category 1 variable

Containment ~~Area~~ Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. 1

~~[For this unit,~~ Containment ~~Area~~ Radiation instrumentation consists of ~~the following:]~~ 3

Insert 5

Insert 5 1 3

two channels. Each channel consists of an ambient radiation detector mounted inside containment transmitting to an indicator in the control room. Both channels are recorded. The containment high range radiation monitors have a range of 1 r/hr to 10^8 r/hr. |R1

BASES

LCO (continued)

10. Pressurizer Level

Water

a Category 1 variable

Pressurizer Level is used to determine whether to terminate safety injection (SI), if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the plant conditions necessary to establish natural circulation in the RCS and to verify that the plant is maintained in a safe shutdown condition. 1 R1

Water

~~[For this unit,~~ Pressurizer Level instrumentation consists of ~~the following:]~~ 3 R1

Insert 6

11. Steam Generator Water Level

a Type A and Category 1 variable

Steam Generator Water Level is provided to monitor operation of decay heat removal via the steam generators. The Category I indication of steam generator level ~~is the extended startup range level instrumentation. The extended startup range level~~ covers a span of ~~6~~ inches to ~~394~~ inches ~~above the lower tubesheet~~. The measured differential pressure is displayed ~~in inches of water at 68°F~~. ~~Temperature compensation of this indication is performed manually by the operator.~~ Redundant monitoring capability is provided by two trains of instrumentation. ~~The uncompensated level signal is input to the plant computer, a control room indicator, and the [Auxiliary Feedwater (AFW)] Control System~~. 1

0

183

as 0 – 100% level. 1

converted and

O

~~At some plants,~~ operator action is based on the control room indication of Steam Generator Water Level. The RCS response during a design basis small break LOCA is dependent on the break size. ~~For a certain range of break sizes, the boiler condenser mode of heat transfer is necessary to remove decay heat. At these plants,~~ ~~extended~~ ~~startup range~~ level is a Type A variable because the operator must manually raise and control the steam generator level to establish ~~boiler condenser heat transfer~~. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated ~~extended startup range~~ level reaches the ~~boiler condenser setpoint~~. 1

Steam Generator Water L

steaming

minimum required level

1

Insert 6

3

two channels. Each channel consists of a differential pressure instrument transmitting to an indicator in the control room. One channel is recorded. The pressurizer water level instrument has a calibration range of 145.6 inches to 331.8 inches (0 – 100% indicated level). |R1

BASES

LCO (continued)

12. Condensate Storage Tank (CST) Level

a Category 1 variable

CST Level is provided to ensure water supply for ~~[~~AFW~~]~~. The CST provides the ensured safety grade water supply for the ~~[~~AFW~~]~~ System. ~~The CST consists of two identical tanks connected by a common outlet header.~~ Inventory is monitored by a 0 to ~~144 inch~~ 50 foot level indication ~~for each tank~~. CST Level is displayed on a control room indicator~~, strip chart recorder, and plant computer~~. In addition, a control room annunciator alarms on low level. 1 3 1

~~At some plants, CST Level is considered a Type A variable because the control room meter and annunciator are considered the primary indication used by the operator. The DBAs that require [AFW] are the loss of electric power, steam line break (SLB), and small break LOCA~~. The CST is the initial source of water for the ~~[~~AFW~~]~~ System. However, as the CST is depleted, manual operator action is necessary to replenish the CST ~~or align suction to the [AFW] pumps from the hotwell~~. 3 3 1

13, 14, 15, 16. Core Exit Temperature

a Category 1 variable

Core Exit Temperature is provided for verification and long term surveillance of core cooling. 1

An evaluation was made of the minimum number of valid core exit thermocouples necessary for inadequate core cooling detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core uncovery and trend the ensuing core heatup. The evaluations account for core nonuniformities including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate or inadequate core cooling detection is ensured with two valid core exit thermocouples per quadrant.

The design of the Incore Instrumentation System includes a Type K (chromel alumel) thermocouple within each of the 56 incore instrument detector assemblies.

LCO (continued)

The junction of each thermocouple is located a few inches above the fuel assembly, inside a structure that supports and shields the incore instrument detector assembly string from flow forces in the outlet plenum region. These core exit thermocouples monitor the temperature of the reactor coolant as it exits the fuel assemblies.

The core exit thermocouples have a usable temperature range from 32°F to 2300°F, although accuracy is reduced at temperatures above 1800°F.

17. ~~[Auxiliary Feedwater (AFW)] Flow~~ Insert 7 3

~~[AFW] Flow is provided to monitor operation of decay heat removal via the steam generators.~~

~~The [AFW] Flow to each steam generator is determined from a differential pressure measurement calibrated to a span of 0 gpm to 1200 gpm. Redundant monitoring capability is provided by two independent trains of instrumentation for each steam generator. Each differential pressure transmitter provides an input to a control room indicator and the plant computer. Since the primary indication used by the operator during an accident is the control room indicator, the PAM instrumentation Specification deals specifically with this portion of the instrument channel.~~

~~At some plants [AFW] Flow is a Type A variable because operator action is required to throttle flow during an SLB accident in order to prevent the [AFW] pumps from operating in runout conditions. [AFW] Flow is also used by the operator to verify that the [AFW] System is delivering the correct flow to each steam generator. However, the primary indication used by the operator to ensure an adequate inventory is steam generator level.~~ 1

Two channels are required to be OPERABLE for all but one Function. Two OPERABLE channels ensure that no single failure, within either the PAM instrumentation or its auxiliary supporting features or power sources (concurrent with the failures that are a condition of or result from a specific accident), prevents the operators from being presented the information necessary for them to determine the safety status of the plant and to bring the plant to and maintain it in a safe condition following that accident.

1

Insert 7

Containment Hydrogen Concentration

Containment Hydrogen Concentration is a Type A and Category 1 variable provided to allow an ability to measure containment hydrogen concentration without laboratory analysis. The channels provide indication over a range of 0 – 10%.

BASES

LCO (continued)

9 In Table 3.3.~~11~~-1 the exception to the two channel requirement is Containment Isolation Valve Position. 2

Two OPERABLE channels of core exit thermocouples are required for each channel in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number ~~and locations~~ provided for diagnosis of local core problems. ~~Therefore, two randomly selected thermocouples may not be sufficient to meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one be located near the center of the core and the other near the core perimeter, such that the pair of core exit thermocouples indicate the radial temperature gradient across their core quadrant. Plant specific evaluations in response to Item II.F.2 of NUREG-0737 should have identified the thermocouple pairings that satisfy these requirements.~~ 1
Two sets of two thermocouples in each quadrant ensure a single failure will not disable the ability to determine the radial temperature gradient.

Core uniformities including in-core effects of the radial decay power distribution, ex-core effects of condensate runback in the hot legs and non-uniform inlet temperatures were accounted for in the evaluation.

For loop and steam generator related variables, the required information is individual loop temperature and individual steam generator level. In these cases two channels are required to be OPERABLE for each loop of steam generator to redundantly provide the necessary information.

In the case of Containment Isolation Valve Position, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of the passive valve or via system boundary status. If a normally active containment isolation valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, plant conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

BASES

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.~~11~~-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function. When the Required Channels in Table 3.3.~~11~~-1 are specified (e.g., on a ~~per steam line,~~ per loop, per SG, etc., basis) then Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

This Required Action specifies initiation of actions in accordance with Specification 5.6.~~5~~, which requires a written report to be submitted to the Nuclear Regulatory Commission. This report discusses the cause of the inoperability and identifies proposed restorative Required Actions. This Required Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Required Actions are identified before a loss of functional capability condition occurs.

C.1

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable

BASES

ACTIONS (continued)

because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

D.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.~~11~~-1. The applicable Condition referenced in the Table is Function dependent. Each time Required Action C.1 is not met, and the associated Completion Time has expired, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition. 9 2

E.1 and E.2

If the Required Action and associated Completion Time of Condition D is not met, and Table 3.3.~~11~~-1 directs entry into Condition E, the plant must be brought to a MODE in which the requirements of this LCO do not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. 9 2

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

~~[At this plant,~~ alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the plant, but rather to follow the directions of Specification 5.6.~~5~~. 4 The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.~~]~~ 3 5

BASES

SURVEILLANCE REQUIREMENTS

A Note at the beginning of the Surveillance Requirements specifies that the following SRs apply to each PAM instrumentation Function in Table 3.3.~~11~~-1. 9 2

SR 3.3.~~11~~.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

~~[The Frequency of 31 days is based upon plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is a rare event. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel during normal operational use of the displays associated with this LCO's required channels.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.3.~~11~~.2

A CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies the channel responds to the measured parameter within the necessary range and accuracy. A Note allows exclusion of neutron detectors from the CHANNEL CALIBRATION.

[~~At this unit,~~ CHANNEL CALIBRATION shall find measurement errors are within the following acceptance criteria: ~~]~~

For the Containment ~~Area~~ Radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr, and a one point calibration check of the detector below 10 R/hr with a gamma source. | R1

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD) sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

[~~The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by an [18] month calibration interval for the determination of the magnitude of equipment drift.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 3

REFERENCES

1. ~~Plant specific document (e.g., FSAR, NRC Regulatory Guide 1.97, SER letter).~~ St. Lucie Unit No. 2 Engineering Evaluation of Instrumentation System for Regulatory Guide 1.97, Rev. 3 (ML17216A337).
2. Regulatory Guide 1.97, , Rev. 3
3. NUREG-0737, Supplement 1.
4. NRC Safety Evaluation Report (SER).

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.9 BASES, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The heading for ISTS 3.3.11 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2) and Chemical and Volume Control System (CVCS) Isolation Signal (ISTS 3.3.9). Therefore, ISTS 3.3.2 and ISTS 3.3.9 are not included in the PSL ITS. The successive Specifications are renumbered as necessary.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Changes have been made to reflect the change to numbering in ISTS Specification 5.6.5.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.9, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 10

3.3.10, REMOTE SHUTDOWN SYSTEM

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

INSTRUMENTATION

REMOTE SHUTDOWN ~~INSTRUMENTATION~~

SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.3.10

~~3.3.3.5~~ The ~~remote shutdown monitoring instrumentation channels shown in Table 3.3-9~~ shall be OPERABLE ~~with readouts displayed external to the control room~~.

System Functions

LA01

LA02

Applicability

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Add proposed ACTIONS Note

A02

Condition A

~~With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either:~~

One or more required Functions inoperable

ACTION A.1

~~a.~~ Restore ~~the inoperable channel~~ to OPERABLE status within 30 days, or

required Functions

Add proposed ACTION B.1

M01

ACTION B.2

~~b.~~ Be in ~~HOT SHUTDOWN~~ within ~~the next~~ 12 hours.

MODE 4

SURVEILLANCE REQUIREMENTS

~~NOTE~~

~~CHANNEL CALIBRATION is not applicable to the reactor trip breaker indication.~~

L01

~~4.3.3.5~~ Each ~~remote shutdown monitoring~~ instrumentation channel that is normally energized shall be demonstrated

SR 3.3.10.1

SR 3.3.10.2

OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION ~~operations~~ in accordance with the Surveillance Frequency Control Program.

required

Add proposed SR 3.3.10.3

M02

LCO 3.3.10

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LA01

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Reactor Trip Breaker Indication	SWGR	OPEN-CLOSE	1/trip breaker
2. Pressurizer Pressure	Hot Shutdown Panel	1500-2500 psia	1
3. Pressurizer Level	Hot Shutdown Panel	0-100%	1
4. Main Steam Pressure	Hot Shutdown Panel	0-1200 psig	1/steam generator
5. Steam Generator Level	Hot Shutdown Panel	0-100%	1/steam generator
6. Cold Leg Temperature	Hot Shutdown Panel	0-600°F	1

~~DELETED~~

~~DELETED~~

A01

~~Pages 3/4 3-38 through 3/4 3-40 (Amendment No. 115) have been deleted from the Technical Specifications. The next page is 3/4 3-41.~~

A01

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM ~~INSTRUMENTATION~~

LIMITING CONDITION FOR OPERATION

LCO 3.3.10 ~~3.3.3.5~~ The ~~remote shutdown system transfer switches, control and instrumentation channels shown in Table 3.3-9~~ Functions shall be OPERABLE. LA01

Applicability **APPLICABILITY:** MODES 1, 2, and 3.

ACTION:

Add proposed ACTIONS Note A02

One or more required Functions inoperable

required Functions

Condition A

~~a.~~ ~~With the number of OPERABLE remote shutdown channels less than the Required Number of Channels shown in Table 3.3-9, either~~ restore ~~the inoperable channel~~ to OPERABLE status within 30 days or be in ~~HOT~~

ACTION A.1

ACTION B.2 MODE 4 ~~SHUTDOWN~~ within ~~the next~~ 12 hours.

Add proposed ACTION B.1 M01

~~b. With the number of OPERABLE remote shutdown channels less than the Minimum Channels OPERABLE requirements of Table 3.3-9, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.~~ L02

SURVEILLANCE REQUIREMENTS

L01

~~**NOTE**~~

~~CHANNEL CALIBRATION is not applicable to reactor trip breaker indication.~~

required that is normally energized

~~4.3.3.5.1~~ Each ~~remote shutdown monitoring~~ instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION ~~operation~~s in accordance with the Surveillance Frequency Control Program.

SR 3.3.10.1

SR 3.3.10.2

Add proposed SR 3.3.10.3 M02

required

SR 3.3.10.4 ~~4.3.3.5.2~~ Each ~~remote shutdown system~~ instrumentation transfer switch and control circuit ~~shall be demonstrated OPERABLE by verifying~~ it~~s~~ capabil~~ity~~ to perform ~~its~~ intended function~~(s)~~ in accordance with the Surveillance Frequency Control Program.

the

LCO 3.3.10

TABLE 3.3-9

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

LA01

L02

INSTRUMENT	READOUT LOCATION	CHANNELS RANGE	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Power Range Neutron Flux	Hot Shutdown Panel	2×10^{-8}% - 200%	2	1
2. Reactor Trip Breaker Indication	Reactor Trip Switch Gear (RB)	OPEN-CLOSE	1/trip breaker	1/trip breaker
3. Reactor Coolant Temperature – T_{Cold}	Hot Shutdown Panel	0ºF - 600ºF	2	1
4. Pressurizer Pressure	Hot Shutdown Panel	0 - 3000 psia	2	1
5. Pressurizer Level	Hot Shutdown Panel	0 - 100% level	2	1
6. Steam Generator Pressure	Hot Shutdown Panel	0 - 1200 psia	1/steam generator	1/steam generator
7. Steam Generator Level	Hot Shutdown Panel	0 - 100% level	2/steam generator	1/steam generator
8. Shutdown Cooling Flow Rate	Hot Shutdown Panel	0 - 5000 gpm	2	1
9. Shutdown Cooling Temperature	Hot Shutdown Panel	0ºF - 350ºF	2	1
10. Diesel Generator Voltage	Hot Shutdown Panel	0 - 5250 V	1/diesel generator	1/diesel generator
11. Diesel Generator Power	Hot Shutdown Panel	0 - 5000 kW	1/diesel generator	1/diesel generator
12. Atmospheric Dump Valve Pressure	Hot Shutdown Panel	0 - 1200 psig	1/steam generator	1/steam generator
13. Charging Flow/Pressure	Hot Shutdown Panel	0 - 150 gpm/ 0 - 3000 psia	2	1
CONTROLS/ISOLATE SWITCHES				
1. Atmospheric Stm Dump Controllers	Hot Shutdown Panel/ RAB431	N.A.	2/steam generator	1/steam generator
2. Aux. Spray Valves	Hot Shutdown Panel/ RAB431	N.A	2	1
3. Charging Pump Controls	Hot Shutdown Panel/ RAB431	N.A	3	2
4. Letdown Isol Valve	Hot Shutdown Panel/ RAB431	N.A	3	2
5. AFW Pump/Valve Controls	Hot Shutdown Panel/ RAB431	N.A	3	2
6. AFW Pump Steam Inlet Valve	Hot Shutdown Panel/ RAB431	N.A	2	1
7. Pzr Heater Controls	Hot Shutdown Panel/ RAB431	N.A	6	3

~~DELETED~~

DISCUSSION OF CHANGES
ITS 3.3.10, REMOTE SHUTDOWN SYSTEM

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.3.3.5 Action a , and Unit 2 CTS 3.3.3.5 Actions a and b, provide the compensatory actions to take when remote shutdown monitoring instrumentation is inoperable. ITS 3.3.10 ACTIONS provide the compensatory actions for inoperable remote shutdown monitoring instrumentation. The ITS 3.3.10 ACTIONS include a Note that allows separate Condition entry for each Function. This modifies the CTS by providing a specific allowance to enter the Action for each inoperable remote shutdown monitoring instrumentation Function. This change is acceptable because it clearly states the current requirement. The CTS considers each remote shutdown monitoring instrumentation Function to be separate and independent from the others. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.3.3.5 Action a requires that if an inoperable channel cannot be restored to OPERABLE status within the allowed completion time, then the unit shall be placed in HOT SHUTDOWN within the next 12 hours. ITS 3.3.10 ACTION B requires, if a required channel cannot be returned to OPERABLE status within the associated Completion Time, then the unit shall be in MODE 3 (HOT STANDBY) within 6 hours and MODE 4 (HOT SHUTDOWN) within 12 hours. This changes the CTS requirements by specifying that MODE 3 must be achieved within 6 hours. The purpose of ITS 3.3.10 Required Action B.1 is to specify consistent Completion Times to shutdown the unit from full power to MODE 3. This change is acceptable because the proposed Completion Time is sufficient to allow an operator to reduce power from full power to MODE 3 in a controlled manner without challenging unit safety systems. The 6 hour time provided to reach MODE 3 is also consistent with the time provided in similar actions in both the CTS and ITS. The change has been designated as more restrictive because it specifies the amount of time within which the unit must be placed in MODE 3.

M02 ITS SR 3.10.3.3 states "Perform CHANNEL FUNCTIONAL TEST of the reactor trip circuit breaker open/closed indication." CTS does not require this Surveillance. This changes CTS by adding a requirement to verify the OPERABILITY of the reactor trip circuit breaker (RTCB) open/closed indication by actuating the RTCBs at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 18 months.

The purpose of ITS SR 3.3.10.3 is to provide assurance that the reactor trip circuit breaker (RTCB) open/closed indication represents the correct position of the RTCBs when open and closed. This change is necessary to ensure sufficient testing is conducted at a frequency to assure that the correct open/closed position of the RTCBs is indicated and that the LCO will be met. This SR verifies the RCTB open/closed indication represents the correct position of the RTCBs when open and closed.

PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). The performance Frequency of ITS SR 3.3.10.3 is in accordance with the SFCP. The initial periodic Frequency established in accordance with the SFCP will be 18 months consistent with the ISTS SR 3.3.12.4 and necessary because the RTCBs cannot be exercised while the unit is at power. Industry operating experience has shown that these components usually pass the Surveillance when performed at a Frequency of once every 18 months. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. See FPL (PSL Unit 1 and Unit 2) "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program" (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066), which approved the adoption of a SFCP at PSL.

This change is designated as more restrictive because a Surveillance Requirement is added to the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.3.3.5 requires the remote shutdown monitoring instrumentation in Table 3.3-9 to be OPERABLE. CTS Table 3.3-9 lists each of the required remote shutdown monitoring instruments, the location of each instrument readout, the measurement range of each instrument, the minimum number of channels required for each instrument, and the minimum channels OPERABLE for each instrument (Unit 2). ITS LCO 3.3.10 states that the remote shutdown monitoring instrumentation Functions shall be OPERABLE. This changes the CTS by moving the details in Tables 3.3-9 from the Technical Specifications to the ITS Bases. Note that Discussion of Change L02 (Unit 2) describes deleting the minimum channels OPERABLE column in Unit 2 Table 3.3-9, while retaining the minimum number of channels required for each instrument requirement.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the remote shutdown monitoring instrumentation to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 **Unit 1 only:** *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.3.3.5 states that the remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE "with readouts displayed external to the control room." ITS LCO 3.3.10 states that the remote shutdown monitoring instrumentation Functions shall be OPERABLE. This changes the CTS by moving the requirement for readouts displayed external to the control room from the Technical Specifications to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the remote shutdown monitoring instrumentation to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 5 – Deletion of Surveillance Requirement)* Unit 1 CTS 4.3.3.5 and Unit 2 CTS 4.3.3.5.1 include a Surveillance Requirements Note that states the CHANNEL CALIBRATION is not applicable to reactor trip breaker indication. ITS 3.3.10 Surveillance Requirements do not provide this note. Rather than a Note, ITS 3.3.10 provides Surveillance SR 3.3.10.3 which requires performing a CHANNEL FUNCTIONAL TEST of the reactor trip circuit breaker (RTCB) open/closed indication. This changes the CTS by deleting the Surveillance Requirements Note and adding ITS SR 3.3.10.3 to perform a CHANNEL FUNCTIONAL TEST of the reactor trip circuit breaker open/closed indication.

The purpose of Unit 1 CTS 4.3.3.5 and Unit 2 CTS 4.3.3.5.1 Surveillance Requirements Note is to clarify that CHANNEL CALIBRATION is not applicable to the reactor trip breaker indication. The purpose of ITS SR 3.3.10.3 is to

perform a CHANNEL FUNCTIONAL TEST. This Surveillance verifies the OPERABILITY of the RTCB open/closed indication by actuating the RTCBs.

This change is acceptable because the deleted Surveillance Requirement Note is not necessary to verify that the remote shutdown reactor trip breaker indication used to meet the LCO can perform its required function. The addition of ITS SR 3.3.10.3 to perform a CHANNEL FUNCTIONAL TEST is appropriate to assure the remote shutdown reactor trip breaker indication continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L02 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* CTS LCO 3.3.3.5 requires the remote shutdown system instrumentation channels shown in Table 3.3.9 to be OPERABLE. CTS Action a requires that with the number of OPERABLE remote shutdown channels less than the required number of channels shown in Table 3.3-9, restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours. CTS Action b requires that with the number of OPERABLE remote shutdown channels less than the Minimum Channels OPERABLE shown in Table 3.3-9, restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours." ITS LCO 3.3.10 states that the Remote Shutdown System Functions shall be OPERABLE. ITS 3.3.10 Required Action A.1 states that with one or more required Functions inoperable, restore the required Functions to OPERABLE status within 30 days. If Required Action A.1 is not completed within the associated Completion Time, ACTION B is entered, and the unit must be in MODE 3 in 6 hours and MODE 4 in 12 hours. ITS 3.3.10 does not contain an Action to restore the inoperable channel to OPERABLE status within 7 days with the number of OPERABLE remote shutdown channels less than the Minimum Channels OPERABLE shown in Table 3.3-9. This changes the Unit 2 CTS by allowing 30 days instead of 7 days for a remote shutdown Function to be inoperable when the number of OPERABLE remote shutdown channels is less than the minimum channels OPERABLE shown in CTS Table 3.3-9 before requiring the unit to shutdown. This changes the CTS by retaining the required number of channels requirement and deleting the minimum channels OPERABLE requirement for each Function in Table 3.3-9. With the change from the 7-day to 30-day Completion Time when a Function does not meet the minimum channel OPERABLE requirement, deleting the minimum channels OPERABLE requirement for each Function in Table 3.3-9 is bounded by the fact that is most cases the required number of channels, which are retained in the CTS, is greater than the minimum required channels. In some cases the required number of channels is the same as the minimum channels OPERABLE. It should be noted that the PSL Unit 1 CTS 3.3.3.5 provides a 30 day allowed outage time when the number of OPERABLE remote shutdown channels is less than the Minimum Channels OPERABLE shown in CTS Table 3.3-9, so this change does not apply to Unit 1.

The proposed change is acceptable considering that the operability of the Remote Shutdown System instrumentation is not a specific assumption of a design basis accident analysis and the proposed 30-day Completion Time

continues to provide a sufficient limit on plant operation if a Remote Shutdown Function is not restored to operable status. As such, the 30-day Completion is acceptable due to the lower safety significance of the Remote Shutdown System instrumentation relative to systems and components necessary to mitigate design basis accidents and considering the low probability of an event occurring within this time that would require the use of the Remote Shutdown System (i.e., control room evacuation). The proposed change provides additional operational flexibility for equipment restoration without significantly affecting the safe operation of the plant. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~12~~ Remote Shutdown System ~~(Analog)~~
10
3

3.3.3.5 LCO 3.3.~~12~~ The Remote Shutdown System Functions shall be OPERABLE.
10
3

Applicability APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
DOC A02 Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.3.3.5 Action a. A. One or more required Functions inoperable.	A.1 Restore required Functions to OPERABLE status.	30 days
DOC M01 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND	6 hours
3.3.3.5 Action b.	B.2 Be in MODE 4.	~~[~~12~~]~~ hours 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.3.3.5 Table 3.3.3-9 DOC LA01 SR 3.3.~~12~~.1 10	~~[~~ Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	~~[31 days~~ 3 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~] 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.3.12.2~~	~~Verify each required control circuit and transfer switch is capable of performing the intended function.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~ 1
SR 3.3.~~12.3~~ 10.2 4.3.3.5 Table 3.3.3-9 DOC LA01	----------NOTE---------- Neutron detectors are excluded from the CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION for each required instrumentation channel.	3 ~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2 2
SR 3.3.~~12.4~~ 10.3 DOC M02	~~[~~ Perform CHANNEL FUNCTIONAL TEST of the reactor trip circuit breaker open/closed indication.	~~[18 months~~ 3 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~ 2 2

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.~~12~~ Remote Shutdown System ~~(Analog)~~

10

3

3.3.3.5 LCO 3.3.~~12~~ The Remote Shutdown System Functions shall be OPERABLE.

10

3

Applicability APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------

DOC A02 Separate Condition entry is allowed for each Function.

	CONDITION		REQUIRED ACTION	COMPLETION TIME
3.3.3.5 Action a.	A. One or more required Functions inoperable.	A.1	Restore required Functions to OPERABLE status.	30 days
DOC M01	B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
		AND		
3.3.3.5 Action b.		B.2	Be in MODE 4.	~~[~~12~~]~~ hours

2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.3.3.5.1 Table 3.3.3-9 DOC LA01	SR 3.3.~~12~~.1	~~[~~ Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

10

3

2

2

3

1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.3.3.5.2 Table 3.3.3-9 DOC LA01	SR 3.3.~~12.2~~ 10.4	Verify each required control circuit and transfer switch is capable of performing the intended function.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 2 2
4.3.3.5.1 Table 3.3.3-9 DOC LA01	SR 3.3.~~12.3~~ 10.2	----------NOTE---------- Neutron detectors are excluded from the CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION for each required instrumentation channel.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 2 2
DOC M02	SR 3.3.~~12.4~~ 10.3	~~[~~ Perform CHANNEL FUNCTIONAL TEST of the reactor trip circuit breaker open/closed indication.	~~[18 months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~	3 2 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.10, REMOTE SHUTDOWN SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS Specification number.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~12~~ Remote Shutdown System ~~(Analog)~~

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the ~~[~~Auxiliary Feedwater (AFW) System~~]~~ and ~~the steam generator safety valves or~~ the steam generator atmospheric dump valves can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the ~~[~~AFW System~~]~~ and the ability to borate the Reactor Coolant System (RCS) from outside the control room allow extended operation in MODE 3.

In the event that the control room becomes inaccessible, the operators can establish control at the ~~remote~~ shutdown panel and place and maintain the unit in MODE 3. ~~Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will be operated locally at the switchgear, motor control panels, or other local stations.~~ The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the Remote Shutdown System control and instrumentation Functions ensures that there is sufficient information available on selected plant parameters to place and maintain the plant in MODE 3, should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the plant in a safe condition in MODE 3.

The criteria governing the design and the specific system requirements of the Remote Shutdown System are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

The Remote Shutdown System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The Remote Shutdown System LCO provides the requirements for the OPERABILITY of the instrumentation ~~and controls~~ necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation ~~and controls~~ required are listed in Table B 3.3.~~12~~-1.

BASES

LCO (continued)

~~The controls, instrumentation, and transfer switches are those required for:~~

- ~~Core Reactivity Control (initial and long term),~~
- ~~RCS Pressure Control,~~
- ~~Decay Heat Removal via the [AFW System] and the safety valves or steam generator ADVs,~~
- ~~RCS Inventory Control via charging flow, and~~
- ~~Safety support systems for the above Functions, as well as service water, component cooling water, and onsite power including the diesel generators.~~

1

A Function of a Remote Shutdown System is OPERABLE if all instrument ~~and control~~ channels needed to support the remote shutdown Functions are OPERABLE. ~~In some cases, Table B 3.3.12-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Function is OPERABLE as long as one channel of any of the alternate information or control sources for each Function is OPERABLE.~~

1

1

The Remote Shutdown System instrumentation ~~and control circuits~~ covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure that the instrument ~~and control circuits~~ will be OPERABLE if plant conditions require that the Remote Shutdown System be placed in operation.

1

1

s

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the unit is already subcritical and in the condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument ~~control~~ Functions if control room instruments ~~or control~~ become unavailable.

1

BASES

ACTIONS

A Remote Shutdown System ~~division~~ is inoperable when ~~each~~ Function ~~is not accomplished by at least one designated Remote Shutdown System channel that satisfies the~~ OPERABILITY criteria for the channel's Function. These criteria are outlined in the LCO section of the Bases.

channel
a
1
does not satisfy the minimum channels

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

A.1

required Functions
1

Condition A addresses the situation where one or more ~~channels~~ of the Remote Shutdown System are inoperable. ~~This includes the control and transfer switches for any required Function.~~
1

required Functions

The Required Action is to restore the ~~divisions~~ to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.
1

B.1 and B.2

If the Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within ~~[~~12~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.
2

SURVEILLANCE REQUIREMENTS

~~[~~ SR 3.3.~~12~~.1
10
2
3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that

BASES

SURVEILLANCE REQUIREMENTS (continued)

the instrumentation continues to operate properly between each CHANNEL CALIBRATION. Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. As specified in the Surveillance, a CHANNEL CHECK is only required for those channels that are normally energized. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

~~[The Frequency of 31 days is based on plant operating experience that demonstrates channel failure is rare.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4

~~]~~

~~SR 3.3.12.2~~ 1

~~SR 3.3.12.2 verifies that each required Remote Shutdown System transfer switch and control circuit performs its intended function. This verification is performed from the reactor shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the plant can be placed and maintained in MODE 3 from the reactor shutdown panel and the local control stations. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an~~ 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~unplanned transient if the Surveillance were performed with the reactor at power. Operating experience demonstrates that Remote Shutdown System control channels seldom fail to pass the Surveillance when performed at a Frequency of once every [18] months.~~ 1

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~ 1

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4

SR 3.3.~~12~~.3 3

10.2

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to the measured parameter within the necessary range and accuracy. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD) sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

~~[The 18 month Frequency is based upon the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4

The SR is modified by a Note, which excludes neutron detectors from the CHANNEL CALIBRATION.

[SR 3.3.~~12.4~~ 10.3 3

SR 3.3.~~12.4~~ is the performance of a CHANNEL FUNCTIONAL TEST. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This Surveillance should verify the OPERABILITY of the reactor trip circuit breaker (RTCB) open/closed indication ~~on the remote shutdown panels~~ by actuating the RTCBs. ~~[The Frequency of 18 months was chosen because the RTCBs cannot be exercised while the unit is at power. Operating experience has shown that these components usually pass the Surveillance when performed at a Frequency of once every 18 months. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ 1 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]]~~ 4

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.

2. NRC Safety Evaluation Report (SER).

10

10

3

Table B 3.3.12-1 (page 1 of 1)
Remote Shutdown System Instrumentation and Controls

1

NOTE
This Table is for illustration purposes only. It does not attempt to encompass every Function used at every unit, but does contain the types of Functions commonly found.

INSERT 1

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF DIVISIONS
1. Reactivity Control	
a. Log Power Neutron Flux	[1]
b. Source Range Neutron Flux	[1]
c. Reactor Trip Circuit Breaker Position	[1 per trip breaker]
d. Manual Reactor Trip	[2]
2. Reactor Coolant System Pressure Control	
a. Pressurizer Pressure or RCS Wide Range Pressure	[1]
b. Pressurizer Power Operated Relief Valve Control and Block Valve Control	[1, controls must be for power operated relief valve and block valves on same line]
3. Decay Heat Removal via Steam Generators	
a. Reactor Coolant\ Hot Leg Temperature	[1 per loop]
b. Reactor Coolant Cold Leg Temperature	[1 per loop]
c. Auxiliary Feedwater Controls	[1]
d. Steam Generator Pressure	[1 per steam generator]
e. Steam Generator Level or Auxiliary Feedwater Flow	[1 per steam generator]
f. Condensate Storage Tank Level	[1]
4. Reactor Coolant System Inventory Control	
a. Pressurizer Level	[1]
b. Reactor Coolant Charging Pump Controls	[1]

1

REVIEWER'S NOTE
The number of channels that fulfill GDC 19 requirements for the number of OPERABLE channels required depends upon the plant's licensing basis as described in the NRC plant specific Safety Evaluation Report (SER) (Ref. 2). Generally, two divisions are required to be OPERABLE. However, only one channel is required if the plant has justified such a design and the NRC's SER accepted the justification.

4

5

1

INSERT 1

	INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	REQUIRED CHANNELS
1.	Reactor Trip Breaker Indication	SWGR	Open-Close	1/trip breaker
2.	Pressurizer Pressure	Hot Shutdown Panel	1500-2500 psia	1
3.	Pressurizer Level	Hot Shutdown Panel	0-100%	1
4.	Steam Generator Pressure	Hot Shutdown Panel	0-1200 psig	1/steam generator
5.	Steam Generator Level	Hot Shutdown Panel	0-100%	1/steam generator
6.	Cold Leg Temperature	Hot Shutdown Panel	0-600°F	1

B 3.3 INSTRUMENTATION ~~(Analog)~~

B 3.3.~~12~~ Remote Shutdown System ~~(Analog)~~

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the ~~[~~Auxiliary Feedwater (AFW) System~~]~~ and ~~the steam generator safety valves or~~ the steam generator atmospheric dump valves can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the ~~[~~AFW System~~]~~ and the ability to borate the Reactor Coolant System (RCS) from outside the control room allow extended operation in MODE 3.

In the event that the control room becomes inaccessible, the operators can establish control at the ~~remote~~ shutdown panel and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located at the ~~remote~~ shutdown panel. Some controls and transfer switches will be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the Remote Shutdown System control and instrumentation Functions ensures that there is sufficient information available on selected plant parameters to place and maintain the plant in MODE 3, should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the plant in a safe condition in MODE 3.

The criteria governing the design and the specific system requirements of the Remote Shutdown System are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

The Remote Shutdown System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The Remote Shutdown System LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation and controls required are listed in Table B 3.3.~~12~~-1.

BASES

LCO (continued)

~~The controls, instrumentation, and transfer switches are those required for:~~

- ~~Core Reactivity Control (initial and long term),~~
- ~~RCS Pressure Control,~~
- ~~Decay Heat Removal via the [AFW System] and the safety valves or steam generator ADVs,~~
- ~~RCS Inventory Control via charging flow, and~~
- ~~Safety support systems for the above Functions, as well as service water, component cooling water, and onsite power including the diesel generators.~~

1

A Function of a Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the remote shutdown Functions are OPERABLE. ~~In some cases, Table B 3.3.12-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Function is OPERABLE as long as one channel of any of the alternate information or control sources for each Function is OPERABLE.~~

1

The Remote Shutdown System instrumentation and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure that the instrument and control circuits will be OPERABLE if plant conditions require that the Remote Shutdown System be placed in operation.

s

1

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the unit is already subcritical and in the condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument control Functions if control room instruments or control become unavailable.

BASES

ACTIONS

does not satisfy the required number of channels or the minimum channels

channel

a

A Remote Shutdown System ~~division~~ is inoperable when ~~each~~ Function ~~is not accomplished by at least one designated Remote Shutdown System channel that satisfies the~~ OPERABILITY criteria for the channel's Function. These criteria are outlined in the LCO section of the Bases. 1

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

A.1

required Functions

Condition A addresses the situation where one or more ~~channels~~ of the Remote Shutdown System are inoperable. This includes the control and transfer switches for any required Function. 1

required Functions

The Required Action is to restore the ~~divisions~~ to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room. 1

B.1 and B.2

If the Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within ~~[~~12~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems. 2

SURVEILLANCE REQUIREMENTS

~~[~~ SR 3.3.~~12~~.1

10

2 3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that

BASES

SURVEILLANCE REQUIREMENTS (continued)

the instrumentation continues to operate properly between each CHANNEL CALIBRATION. Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. As specified in the Surveillance, a CHANNEL CHECK is only required for those channels that are normally energized. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

~~[The Frequency of 31 days is based on plant operating experience that demonstrates channel failure is rare.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4

~~]~~

SR 3.3.~~12.2~~ 10.4 3

SR 3.3.~~12.2~~ verifies that each required Remote Shutdown System transfer switch and control circuit performs its intended function. This verification is performed from the reactor shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the plant can be placed and maintained in MODE 3 from the reactor shutdown panel and the local control stations. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~unplanned transient if the Surveillance were performed with the reactor at power. Operating experience demonstrates that Remote Shutdown System control channels seldom fail to pass the Surveillance when performed at a Frequency of once every [18] months.~~ 2

~~OR~~

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~~----------]~~

SR 3.3.~~12~~.3 3
10.2

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to the measured parameter within the necessary range and accuracy. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD) sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

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~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4

The SR is modified by a Note, which excludes neutron detectors from the CHANNEL CALIBRATION.

[SR 3.3.~~12.4~~ 10.3 3

SR 3.3.~~12.4~~ is the performance of a CHANNEL FUNCTIONAL TEST. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This Surveillance should verify the OPERABILITY of the reactor trip circuit breaker (RTCB) open/closed indication ~~on the remote shutdown panels~~ by actuating the RTCBs. ~~[The Frequency of 18 months was chosen because the RTCBs cannot be exercised while the unit is at power. Operating experience has shown that these components usually pass the Surveillance when performed at a Frequency of once every 18 months. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ 1 2

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~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]]~~ 4

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.
2. NRC Safety Evaluation Report (SER).

10

10

3

Table B 3.3.12-1 (page 1 of 1)
Remote Shutdown System Instrumentation and Controls

1

NOTE
This Table is for illustration purposes only. It does not attempt to encompass every Function used at every unit, but does contain the types of Functions commonly found.

INSERT 1

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF DIVISIONS
1. Reactivity Control	
a. Log Power Neutron Flux	[1]
b. Source Range Neutron Flux	[1]
c. Reactor Trip Circuit Breaker Position	[1 per trip breaker]
d. Manual Reactor Trip	[2]
2. Reactor Coolant System Pressure Control	
a. Pressurizer Pressure or RCS Wide Range Pressure	[1]
b. Pressurizer Power Operated Relief Valve Control and Block Valve Control	[1, controls must be for power operated relief valve and block valves on same line]
3. Decay Heat Removal via Steam Generators	
a. Reactor Coolant\ Hot Leg Temperature	[1 per loop]
b. Reactor Coolant Cold Leg Temperature	[1 per loop]
c. Auxiliary Feedwater Controls	[1]
d. Steam Generator Pressure	[1 per steam generator]
e. Steam Generator Level or Auxiliary Feedwater Flow	[1 per steam generator]
f. Condensate Storage Tank Level	[1]
4. Reactor Coolant System Inventory Control	
a. Pressurizer Level	[1]
b. Reactor Coolant Charging Pump Controls	[1]

1

REVIEWER'S NOTE
The number of channels that fulfill GDC 19 requirements for the number of OPERABLE channels required depends upon the plant's licensing basis as described in the NRC plant specific Safety Evaluation Report (SER) (Ref. 2). Generally, two divisions are required to be OPERABLE. However, only one channel is required if the plant has justified such a design and the NRC's SER accepted the justification.

4

1

INSERT 1

	INSTRUMENT	LOCATION	RANGE	REQUIRED CHANNELS
1.	Power Range Neutron Flux	Hot Shutdown Panel	2 x 10^{-8}% - 200%	2
2.	Reactor Trip Breaker Indication	Reactor Trip Switch Gear (RB)	OPEN-CLOSE	1/trip breaker
3.	Reactor Coolant Temperature – T_{Cold}	Hot Shutdown Panel	0°F - 600°F	2
4.	Pressurizer Pressure	Hot Shutdown Panel	0 - 3000 psia	2
5.	Pressurizer Level	Hot Shutdown Panel	0 - 100% level	2
6.	Steam Generator Pressure	Hot Shutdown Panel	0 - 1200 psia	1/steam generator
7.	Steam Generator Level	Hot Shutdown Panel	0 - 100% level	2/steam generator
8.	Shutdown Cooling Flow Rate	Hot Shutdown Panel	0 - 5000 gpm	2
9.	Shutdown Cooling Temperature	Hot Shutdown Panel	0°F - 350°F	2
10.	Diesel Generator Voltage	Hot Shutdown Panel	0 - 5250 V	1/diesel generator
11.	Diesel Generator Power	Hot Shutdown Panel	0 - 5000 kW	1/diesel generator
12.	Atmospheric Dump Valve Pressure	Hot Shutdown Panel	0 - 1200 psig	1/steam generator
13.	Charging Flow/Pressure	Hot Shutdown Panel	0 - 150 gpm/ 0 - 3000 psia	2
	CONTROLS / ISOLATE SWITCHES			
1.	Atmospheric Stm Dump Controllers	Hot Shutdown Panel/ RAB431	Not Applicable	2/steam generator
2.	Aux. Spray Valves	Hot Shutdown Panel/ RAB431	Not Applicable	2
3.	Charging Pump Controls	Hot Shutdown Panel/ RAB431	Not Applicable	3
4.	Letdown Isol Valve	Hot Shutdown Panel/ RAB431	Not Applicable	3
5.	AFW Pump/Valve Controls	Hot Shutdown Panel/ RAB431	Not Applicable	3
6.	AFW Pump Steam Inlet Valve	Hot Shutdown Panel/ RAB431	Not Applicable	2
7.	Pzr Heater Controls	Hot Shutdown Panel/ RAB431	Not Applicable	6

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.10, BASES, REMOTE SUTDOWN SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS Specification number and title.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. **Unit 1 only:** The ITS maintains one required channel for each Remote shutdown System Function consistent with the equivalent CTS requirement and licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.10, REMOTE SHUTDOWN SYSTEM

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 11

3.3.11, Logarithmic Neutron Flux Monitoring

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3/4.3 INSTRUMENTATION

Logarithmic Neutron Flux Monitoring

3/4.3.1 ~~REACTOR PROTECTIVE INSTRUMENTATION~~

LIMITING CONDITION FOR OPERATION

LCO 3.3.11

3.3.1.1 ~~As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1~~ shall be OPERABLE.

Two logarithmic neutron flux monitoring channels

A02

Applicability

APPLICABILITY: ~~As shown in Table 3.3-1.~~

MODES 3, 4, and 5.

ACTION:

ACTIONS

As shown in ~~Table 3.3-1~~.

ACTION A

Add proposed SR 3.3.11.2

L01

SURVEILLANCE REQUIREMENTS

required

SR 3.3.11.1

4.3.1.1.1 Each ~~reactor protective instrumentation~~ channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION ~~and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-1~~.

in accordance with the Surveillance Frequency Control Program

L01

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing. Each test shall include at least one channel per function.

See ITS 3.3.1

~~TABLE 3.3-1 (Continued)~~

~~REACTOR PROTECTIVE INSTRUMENTATION~~

Logarithmic Neutron Flux Monitoring

REQUIRED

~~FUNCTIONAL UNIT~~	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ CHANNELS OPERABLE	APPLICABLE MODES	ACTION	LA01
~~11.~~ ~~Wide Range~~ Logarithmic Neutron Flux Monitor						
a. Startup and Operating -- Rate of Change of Power – High	4	2(d)	3	1**, 2 and *	2	See ITS 3.3.1
~~b. Shutdown~~	~~4~~ ← 6	~~0~~	2	3, 4, 5	~~3~~ ← A	
12. Reactor Protection System Logic	4	2	4	1, 2*	4	See ITS 3.3.2
13. Reactor Trip Breakers	4	2	4	1, 2*	4	

LCO 3.3.11 Applicability ACTION A

L02

LA01

~~TABLE 3.3-1 (Continued)~~

~~ACTION STATEMENTS~~

b. Within one hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on than channel provided the other inoperable channel is placed in the tripped condition.

See ITS 3.3.1

Add proposed Required Action A.1 and Note

M01

Required Action A.2

~~ACTION 3~~ - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1~~.1 or 3.1.1.2, as applicable,~~ within ~~1 hour~~ and at least once per 12 hours thereafter.

4 hours

L03

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

See ITS 3.3.2

ITS

A01

ITS 3.3.11

~~TABLE 4.3-1~~

~~REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS~~

Logarithmic Neutron Flux Monitoring

~~FUNCTIONAL UNIT~~	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED	
1. Manual Reactor Trip	N/A	N.A.	S/U(1)	N/A	See ITS 3.3.2
2. Power Level – High					
a. Nuclear Power	SFCP	SFCP(2), SFCP(3), SFCP(5)	SFCP	1,2	
b. ΔT Power	SFCP	SFCP(4), SFCP	SFCP	1	
3. Reactor Coolant Flow – Low	SFCP	SFCP	SFCP	1, 2	
4. Pressurizer Pressure – High	SFCP	SFCP	SFCP	1, 2	
5. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2	
6. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2	See ITS 3.3.1
7. Steam Generator Water Level – Low	SFCP	SFCP	SFCP(6, 7)	1, 2	
8. Local Power Density – High	SFCP	SFCP	SFCP	1	
9. Thermal Margin/Low Pressure	SFCP	SFCP	SFCP	1, 2	
9a. Steam Generator Pressure Difference – High	SFCP	SFCP	SFCP	1, 2	
10. Loss of Turbine -- Hydraulic Fluid Pressure – Low	N.A.	N.A.	S/U(1)	N.A.	
SR 3.3.11.1 SR 3.3.11.2 11. ~~Wide Range~~ Logarithmic Neutron Flux Monitor	SFCP	~~N.A.~~ SFCP L01	~~S/U(1)~~ L01	1, 2, 3, 4, 5 and *	
12. Reactor Protection System Logic	N.A.	N.A.	SFCP and S/U(1)	1, 2 and *	See ITS 3.3.2
13. Reactor Trip Breakers	N.A.	N.A.	SFCP	1, 2 and *	

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breaker closed.

See ITS 3.3.2

~~(1) - If not performed in previous 7 days.~~

L01

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Pwr – ΔT Pwr." During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to ≤ 90% of the maximum allowed THERMAL POWER level with the existing Reactor Coolant Pump combination.

(4) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

(5) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(6) - If the as-found setpoint is either outside its predefined as-found acceptance criteria band or is not conservative with respect to the Allowable Value, then the channel shall be declared inoperable and shall be evaluated to verify that it is functioning as required before returning the channel to service.

(7) - The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Field Trip Setpoint, otherwise that channel shall not be returned to OPERABLE status. The Field Trip Setpoint and the methodology used to determine the Field Trip Setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in the UFSAR Section 7.2.

See ITS 3.3.1

3/4.3 INSTRUMENTATION

Logarithmic Neutron Flux Monitoring

3/4.3.1 ~~REACTOR PROTECTIVE INSTRUMENTATION~~

LIMITING CONDITION FOR OPERATION

LCO 3.3.11

3.3.1 ~~As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1~~ shall be OPERABLE.

Two logarithmic neutron flux monitoring channels

A02

Applicability

APPLICABILITY: ~~As shown in Table 3.3-1~~.

MODES 3, 4, and 5.

ACTION:

ACTIONS

As shown in ~~Table 3.3-1~~.

ACTION A

SURVEILLANCE REQUIREMENTS

required

SR 3.3.11.1
SR 3.3.11.2

4.3.1.1 Each ~~reactor protective instrumentation~~ channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION ~~and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1~~.

in accordance with the Surveillance Frequency Control Program

A03

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing. Each test shall include at least one channel per function.

See ITS 3.3.1

~~TABLE 3.3-1~~

~~REACTOR PROTECTIVE INSTRUMENTATION~~

Logarithmic Neutron Flux Monitoring

REQUIRED

~~FUNCTIONAL UNIT~~	~~TOTAL NO. OF CHANNELS~~	~~CHANNELS TO TRIP~~	~~MINIMUM~~ CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	4 4	~~2~~ ~~2~~	4 4	1, 2 3*, 4*, 5*	1 5
2. Variable Power Level – High	4	2(a)(d)	3	1, 2	2
3. Pressurizer Pressure – High	4	2	3	1, 2	2
4. Thermal Margin/Low Pressure	4	2(a)(d)	3	1, 2	2
5. Containment Pressure – High	4	2	3	1, 2	2
6. Steam Generator Pressure – Low	4/SG	2/SG(b)	3/SG	1, 2	2
7. Steam Generator Pressure Difference – High	4	2(a)(d)	3	1, 2	2
8. Steam Generator Level – Low	4/SG	2/SG	3/SG	1, 2	2
9. Local Power Density – High	4	2(c)(d)	3	1	2
10. Loss of Component Cooling Water to Reactor Coolant Pumps	4	2	3	1, 2	2
11. Reactor Protection System Logic	4	2	3	1, 2 3*, 4*, 5*	2 5
12. Reactor Trip Breakers	4	2(f)	4	1, 2 3*, 4*, 5*	4 5
13. ~~Wide Range~~ Logarithmic Neutron Flux Monitor					
a. Startup and Operating – Rate of Change of Power – High	4	2(e)(g)	3	1**, 2	2
b. ~~Shutdown~~	~~4~~	~~0~~	2	3, 4, 5	~~3~~
14. Reactor Coolant Flow – Low	4/SG	2/SG(a)(d)	3/SG	1, 2	2
15. Loss of Load (Turbine Hydraulic Fluid Pressure – Low)	4	2(c)	3	1	2

LA01

See ITS 3.3.2

See ITS 3.3.1

See ITS 3.3.2

See ITS 3.3.1

See ITS 3.3.1

LCO 3.3.11 Applicability ACTION A

8

A

L02

LA01

~~TABLE 3.3-1 (Continued)~~

~~ACTION STATEMENTS~~

See ITS 3.3.1

ACTION 2 - (Continued)

6.	Cold Leg Temperature	Variable Power Level – High (RPS) Thermal Margin/Low Pressure (RPS) Local Power Density – High (RPS)
7.	Hot Leg Temperature	Variable Power Level – High (RPS) Thermal Margin /Low Pressure (RPS) Local Power Density – High (RPS)

ACTION A

~~ACTION 3~~ - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes*. Verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1~~.1 or 3.1.1.2, as applicable,~~ within ~~1 hour~~ and at least once per 12 hours thereafter.

4 hours

L03

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour, provided the trip breakers of any inoperable channel are in the tripped condition, for surveillance testing per Specification 4.3.1.1.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

Required Action A.1 NOTE

* Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

See ITS 3.3.2

~~TABLE 4.3-1 (Continued)~~

Logarithmic Neutron Flux Monitoring

~~REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS~~

	FUNCTION~~AL UNIT~~	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
12.	Reactor Trip Breakers	N.A.	N.A.	S/U(1), SFCP, SFCP(6)	1, 2, 3*, 4*, 5*	See ITS 3.3.2
13.	~~Wide Range~~ Logarithmic Neutron Flux Monitor	SFCP	SFCP	S/U(1), ~~SFCP~~	1, 2, 3, 4, 5	SR 3.3.11.1 SR 3.3.11.2
14.	Reactor Coolant Flow – Low	SFCP	SFCP	SFCP	1, 2	See ITS 3.3.1
15.	Loss of Load (Turbine Hydraulic Fluid Pressure – Low)	SFCP	N.A.	SFCP	1	

Add proposed ITS SR 3.3.11.2 Note

L04

DISCUSSION OF CHANGES
ITS 3.3.11, LOGARITHMIC NEUTRON FLUX MONITORING

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.3.1.1 and Unit 2 CTS 3.3.1 specify requirements for reactor protective instrumentation and list the requirements for each functional unit in CTS Tables 3.3-1 and 4.3-1, including the Wide Range Logarithmic Neutron Flux Monitor – Shutdown instrumentation. ITS 3.3.11 provides requirements for Logarithmic Neutron Flux Monitoring and provides the requirements in the LCO, Applicability, ACTIONS, and Surveillance Requirements. This changes the CTS by presenting the instrument requirements without use of an instrument table. This change is acceptable as it results solely from the change in the format and presentation of the CTS necessary to conform to the ISTS. As the proposed change is the result of changes in the format and presentation of the CTS requirements, it is designated administrative.

A03 **Unit 2 only:** CTS 4.3.1.1 and Table 4.3-1, Functional Unit 13 (Wide Range Logarithmic Neutron Flux Monitor) require a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and a CHANNEL CALIBRATION on a Frequency in accordance with the Surveillance Frequency Control Program (SFCP). ITS 3.3.11 only requires a CHANNEL CHECK and CHANNEL CALIBRATION and incorporates the CHANNEL FUNCTIONAL TEST into the CHANNEL CALIBRATION. This changes the CTS by incorporating the CHANNEL FUNCTIONAL TEST into the CHANNEL CALIBRATION requirement for the logarithmic neutron flux monitoring channels.

The purpose of the CHANNEL FUNCTIONAL TEST is to ensure components within the instrumentation circuitry are tested more frequently than a CHANNEL CALIBRATION due to the maximum calculated uncertainty of these components. The Frequency in accordance with the SFCP associated with the logarithmic neutron flux monitoring channels is 18 months for both the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION. The CTS 1.4 (ITS Section 1.1) definition for CHANNEL CALIBRATION states, in part: “The CHANNEL CALIBRATION shall encompass the entire channel… and shall include the CHANNEL FUNCTIONAL TEST.” Since the CHANNEL CALIBRATION encompasses the CHANNEL FUNCTIONAL TEST and the periodic Frequency for these Surveillances is equivalent when shutdown, it is unnecessary to present the CHANNEL FUNCTIONAL TEST as a separate test. This change is acceptable as it results solely in a format and presentation change to the CTS by incorporating the CHANNEL FUNCTIONAL TEST into the CHANNEL CALIBRATION. Both Surveillance Requirements are maintained in the ITS via the CHANNEL CALIBRATION (ITS SR 3.3.11.2). As the proposed change is the

result of a change in the format and presentation of the CTS requirements and not a technical change, it is designated administrative.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS Table 3.3-1 Action 3 provides actions for the Wide Range Logarithmic Neutron Flux Monitor – Shutdown Functional Unit (Functional Unit 11.b). Table 3.3-1, Action 3 requires verification of SHUTDOWN MARGIN (SDM) when the number of channels OPERABLE is one less than the required Minimum Channels Operable. ITS 3.3.11 ACTION A requires a similar action but also includes an additional action to immediately suspend all operations involving positive reactivity additions (Required Action A.1). The action includes a Note to allow limited plant cooldown or boron dilution activities provided the change is accounted for in the calculated SDM. This changes the CTS by adding an additional action when one or more required logarithmic neutron flux monitoring channels is inoperable when shutdown (i.e., MODES 3, 4, and 5).

The purpose of the additional required action is to eliminate many of the means by which SDM can be reduced, while allowing normal plant control operations (e.g., Reactor Coolant System inventory management and temperature control). This action is necessary because with one or more required logarithmic neutron flux monitoring channels inoperable, the ability to detect a boron dilution event is degraded. This change is designated as more restrictive because it provides an additional action requirement that is not required by the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.3-1 for reactor protective instrumentation has three columns stating various requirements for each function. These columns are labeled, "TOTAL NO. OF CHANNELS," "CHANNELS TO TRIP," and "MINIMUM CHANNELS OPERABLE." ITS Table 3.3.11 does not retain the "TOTAL NO. OF CHANNELS" or "CHANNELS TO TRIP" columns. This changes the CTS by moving the information of the "TOTAL NO. OF CHANNELS" and "CHANNELS TO TRIP" columns to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the number of required channels and the appropriate Condition to enter if a required channel becomes inoperable. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program

in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 **Unit 1 only:** (*Category 7- Relaxation of Surveillance Frequency*) CTS 4.3.1.1.1 and CTS Table 4.3-1 Functional Unit 11 (Wide Range Logarithmic Neutron Flux Monitor), require performance of a CHANNEL FUNCTIONAL TEST during startup if not performed in the previous 7 days in accordance with Table 4.3-1 Note (1). CTS does not require a CHANNEL CALIBRATION for the logarithmic neutron flux monitor channels. Performing the CHANNEL FUNCTIONAL TEST during startup if not performed in the previous 7 days would routinely equate to every 18 months if there was no other plant startup between refueling outages. The ITS does not require a CHANNEL FUNCTIONAL TEST "during startup if not performed in the previous 7 days," but rather, ITS SR 3.3.11.2 requires a CHANNEL CALIBRATION, which encompasses a CHANNEL FUNCTIONAL TEST, to be performed on a Frequency in accordance with the Surveillance Frequency Control Program (SFCP). A Note is included stating that the neutron detectors are excluded from the CHANNEL CALIBRATION. This changes the CTS by replacing the requirement to perform the CHANNEL FUNCTIONAL TEST during startup (routinely every 18 months) if not performed in the previous 7 days with a requirement to perform a CHANNEL CALIBRATION on a periodic frequency in accordance with the SFCP with an initial periodic Frequency of 18 months.

The purpose of the CTS Table 4.3-1 Functional Unit 11 CHANNEL FUNCTIONAL TEST is to ensure the logarithmic neutron flux monitoring channels are functioning properly. This change is acceptable because the CTS 1.4 (ITS Section) definition of a CHANNEL CALIBRATION encompasses a CHANNEL FUNCTIONAL TEST and also includes a complete check of the instrument channel and will continue to ensure the channels are functioning properly. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. The Note excluding the neutron detectors from the calibration is acceptable because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. CTS 4.0.4 (ITS SR 3.0.4) requires the normal periodic Surveillances to be performed and be current prior to entry into the applicable MODES. Once the applicable MODES are entered, the normal, periodic Surveillance Frequency provides adequate assurance of OPERABILITY. Performing the CHANNEL FUNCTIONAL TEST during startup if not performed in the previous 7 days, as specified in CTS Table 4.3-1 and Note (1) for Functional Unit 11, would routinely be every 18 months if there were no other plant startups between refueling outages. Therefore, replacing the specific Frequency of prior to startup if not performed in the previous 7 days with a specified Frequency of in accordance with the Surveillance Frequency Control Program with an initial periodic frequency of 18 months is considered acceptable.

Requiring a CHANNEL CALIBRATION instead of a CHANNEL FUNCTIONAL TEST is also acceptable because the CHANNEL CALIBRATION encompasses the CHANNEL FUNCTIONAL TEST as defined in Section 1.1 of the ITS.

PSL controls periodic Frequencies for Surveillances in accordance with the SFCP per CTS 6.8.4.o. Therefore, the initial periodic Frequency for SR 3.3.11.2 in accordance with the SFCP will be 18 months consistent with the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION testing frequencies for PSL Unit 2 CTS 4.3.1.1 and Table 4.3-1, Functional Unit 13; and ISTS SR 3.3.13.3. The periodic Frequency is acceptable based upon the assumption of a 22.5 month calibration interval (18 months plus 25% grace) for the determination of the magnitude of equipment drift and is consistent with the CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST Frequencies for the PSL Unit 2 wide range neutron monitoring instrumentation.

The SFCP was established as described in FPL (PSL Unit 1 and Unit 2) "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program" (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066). This change is designated as less restrictive because the Surveillance will be performed less frequently under the ITS than under the CTS.

L02 (*Category 1 – Relaxation of LCO Requirements*) Unit 1 CTS 3.3.1.1 and Unit 2 CTS 3.3.1 specify requirements for reactor protective instrumentation and list the requirements for each functional unit in CTS Tables 3.3-1 and 4.3-1, including the Wide Range Logarithmic Neutron Flux Monitor – Shutdown instrumentation. CTS Table 3.3-1 lists a total of 4 channels and requires a minimum of 2 channels to be OPERABLE for the Wide Range Logarithmic Neutron Flux Monitor – Shutdown Function (Functional Unit 11.b (Unit 1) and 13.b (Unit 2)). TS 3.3.11 provides requirements for Logarithmic Neutron Flux Monitoring and also requires two channels neutron flux monitoring channels to be OPERABLE when shutdown. The ITS 3.3.11 Bases describes the channels required to meet the LCO and states that the excore wide range logarithmic monitoring channels (i.e., post accident and Appendix R monitors) can be substituted for the wide range logarithmic power range channels. Additionally for Unit 2, the startup logarithmic monitoring channels can be substituted for the wide range logarithmic power range channels. This changes the CTS by explicitly stating in the ITS Bases that instruments other than the wide range logarithmic power range monitoring instruments can be used to meet the LCO. This change results in a total of six logarithmic neutron flux monitoring channels that may be used to meet the LCO for Unit 1 and a total of eight logarithmic neutron flux monitoring channels that may be used to meet the LCO for Unit 2.

The purpose of requiring the wide range logarithmic neutron flux monitors to be OPERABLE during MODES 3, 4, and 5 is to alert the operator of reactivity transients (e.g., boron dilution event) initiated from conditions in which the Reactor Protective System is not required. For Unit 1, the excore wide range

logarithmic monitoring channels provide input into the boron dilution alarm and have a wide range of 2 X 10^{-8}% to 200%, which is equivalent to the wide range logarithmic power range channels, and a source range of 1 to 10^5 cps. For Unit 2, the excore wide range logarithmic monitoring channels also have a wide range of 2 X 10^{-8}% to 200%, which is equivalent to the wide range logarithmic power range channels, and a source range of 1 to 10^5 cps. However, separate startup logarithmic monitoring channels provide input to the boron dilution alarm for Unit 2. The startup logarithmic monitoring instruments have a source range of 1 to 10^5 cps. Specific neutron flux monitoring instruments are not assumed in shutdown accidents and transients. However, the timing for operators to take action to mitigate a boron dilution accident is evaluated for the shutdown modes and is based, in part, on alarms and indication of a reactivity increase. This change is acceptable because the excore wide range logarithmic monitoring channels have an instrument range equivalent to the wide range logarithmic power range channels and the Unit 1 excore wide range logarithmic monitoring channels and Unit 2 startup logarithmic monitoring channels provide input to the boron dilution alarm. Therefore, these logarithmic neutron flux monitoring instruments can also alert the operator of reactivity transients, including a boron dilution event, in MODES 3, 4, and 5. This change is designated as less restrictive because less stringent LCO requirements are being applied in ITS than were applied in the CTS.

L03 (*Category 3 - Relaxation of Completion Time*) CTS Table 3.3-1 Action 3 provides compensatory actions for the Wide Range Logarithmic Neutron Flux Monitor – Shutdown Functional Unit when the number of OPERABLE channels is less than the number of minimum required channels Operable. Action 3 requires verification of compliance with Shutdown Margin requirements within 1 hour and at least once per 12 hours thereafter. ITS Required Action A.2 specifies an equivalent action with a Completion Time of 4 hours and once per 12 hours thereafter. This changes the CTS by extending the time to initially verify SHUTDOWN MARGIN (SDM) from 1 hour to 4 hours.

The purpose of the action to verify compliance with SDM is to detect an inadvertent boron dilution of the Reactor Coolant System (RCS) providing operators with time to take action prior to a loss of SDM when the one or more required logarithmic neutron flux monitoring channels is inoperable. The proposed change is acceptable because the initial Completion Time of 4 hours to perform SDM verification takes into consideration that ITS 3.3.11, Required Action A.1 eliminates many of the means by which SDM can be reduced. In addition, the Completion Time extension is justified based on the low likelihood of an event occurring within the allowed Completion Time and the fact that plant conditions will change slowly. The proposed change is designated less restrictive as it results in additional time being allowed to perform a required action.

L04 **Unit 2 only:** *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.3.1.1 and Table 4.3-1, Functional Unit 13 (Wide Range Logarithmic Neutron Flux Monitor) require, in part, a CHANNEL CALIBRATION on a Frequency in accordance with the Surveillance Frequency Control Program (SFCP). ITS SR 3.3.11.2 also requires a CHANNEL CALIBRATION but is modified by a Note stating, “Neutron detectors are excluded from CHANNEL

CALIBRATION.". This changes the CTS by excluding neutron detectors from the CHANNEL CALIBRATION requirement for the logarithmic neutron flux monitoring channels.

The purpose of a CHANNEL CALIBRATION is to ensure that the channel responds within the necessary range and accuracy to known values of the parameter that the channel monitors. Thus, to perform a channel calibration of a neutron flux channel would require including the neutron flux detector in the calibration. Inclusion of neutron flux detectors in the CHANNEL CALIBRATION process is impractical in power reactor applications because to do so would require subjecting the detectors to known neutron fluxes. Neutron detectors are typically excluded from CHANNEL CALIBRATIONS because of the hazards associated with exposing the neutron detectors; for example, CTS Table 4.3-1 Note (4) excludes the detectors of Functional Unit 2 (Variable Power Level High) from CHANNEL CALIBRATION. The logarithmic neutron flux monitoring channels also use neutron detectors, which are subject to the same hazards. The Note excluding the neutron detectors from the calibration is acceptable because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. This proposed change is consistent with historical and current NRC staff requirements as reflected in ISTS. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

~~[~~Logarithmic~~]~~ ~~Power~~ Neutron Flux Monitoring ~~Channels (Analog)~~
3.3.1~~3~~ 1 2 3

CTS

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.1~~3~~ 1 ~~[~~Logarithmic~~]~~ ~~Power~~ Neutron Flux Monitoring ~~Channels (Analog)~~ 1 2 3

Table 3.3-1 Functional Unit 11.a DOC L02

LCO 3.3.1~~3~~ 1 Two ~~channels of [~~logarithmic~~]~~ ~~power level monitoring instrumentation~~ neutron flux monitoring channels shall be OPERABLE. 1 2 3

Applicability

APPLICABILITY: MODES 3, 4, and 5~~, with the reactor trip circuit breakers open or Control Element Assembly (CEA) Drive System not capable of CEA withdrawal~~. 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01 A. One or more required channel(s) inoperable.	A.1 ----------NOTE---------- Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SDM. ---------- Suspend all operations involving positive reactivity additions.	Immediately
	AND	
Table 3.3-1 Action 3 DOC L03	A.2 Perform SDM verification in accordance with SR 3.1.1.1.	4 hours AND Once per 12 hours thereafter

CTS

Neutron Flux

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
Table 4.3-1 Functional Unit 11	SR 3.3.13.1	Perform CHANNEL CHECK.	[12 hours OR In accordance with the Surveillance Frequency Control Program]
DOC L01	SR 3.3.13.2	Perform CHANNEL FUNCTIONAL TEST.	[[92] days OR In accordance with the Surveillance Frequency Control Program]
DOC L01	SR 3.3.13.3	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	[[18] months OR In accordance with the Surveillance Frequency Control Program]

CTS

[Logarithmic] ~~Power~~ Monitoring ~~Channels (Analog)~~ 3.3.1~~3~~

Neutron Flux

3.3 INSTRUMENTATION ~~(Analog)~~

3.3.1~~3~~ [Logarithmic] ~~Power~~ Monitoring ~~Channels (Analog)~~

Neutron Flux

Table 3.3-1 Functional Unit 13.b DOC L02

LCO 3.3.1~~3~~ Two ~~channels of [~~logarithmic] ~~power level monitoring instrumentation~~ shall be OPERABLE.

neutron flux monitoring channels

Applicability

APPLICABILITY: MODES 3, 4, and 5~~, with the reactor trip circuit breakers open or Control Element Assembly (CEA) Drive System not capable of CEA withdrawa~~l.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.3-1 Action 3 A. One or more required channel(s) inoperable.	A.1 ----------NOTE---------- Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SDM. ---------- Suspend all operations involving positive reactivity additions.	Immediately
	AND	
DOC L03	A.2 Perform SDM verification in accordance with SR 3.1.1.1.	4 hours AND Once per 12 hours thereafter

St. Lucie Unit 2

Amendment XXX

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1~~3~~.1	Perform CHANNEL CHECK.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
~~SR 3.3.13.2~~	~~Perform CHANNEL FUNCTIONAL TEST.~~	~~[[92] days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~
SR 3.3.1~~3.3~~	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ----------	
	Perform CHANNEL CALIBRATION.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

Table 4.3-1 Functional Unit 13

DOC A03

DOC L04

Table 4.3-1 Functional Unit 13 DOC A03

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.11, LOGARITHMIC NEUTRON FLUX MONITORING

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The heading for ISTS 3.3.11 includes the parenthetical expression (Analog). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Reactor Protective System (RPS) Instrumentation – Shutdown (ISTS 3.3.2) and Chemical and Volume Control System (CVCS) Isolation Signal (ISTS 3.3.9). Therefore, ISTS 3.3.2 and ISTS 3.3.9 are not included in the PSL ITS. The successive Specifications are renumbered as necessary.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

4. The specified condition in the Applicability of ISTS 3.3.13 state states, “with the reactor trip circuit breakers open or Control Element Assembly (CEA) Drive System not capable of CEA withdrawal,” is not included in ITS 3.3.11. The ITS Applicability of MODES 3, 4, and 5 is consistent with the Applicable Mode in PSL CTS Table 3.3-1 for Functional Unit 11.b (Unit 1) and 13.b (Unit 2) and ensures capability to detect a boron dilution event while shutdown is available regardless of the status of the reactor trip circuit breakers.

5. ISTS 3.3.13.2 is not included in the PSL ITS. In the PSL Unit 2 CTS, the Frequency for the CHANNEL FUNCTIONAL TEST and the CHANNEL CALIBRATION is 18 months in accordance with the Surveillance Frequency Control Program. The definition for CHANNEL CALIBRATION in Section 1.1 states, in part, that the CHANNEL CALIBRATION encompasses the CHANNEL FUNCTIONAL TEST. Therefore, it is unnecessary to provide two separate requirements. This change is also made to PSL Unit 1 to be consistent with PSL Unit 2 since the instrumentation calibration tolerance is the same for both units. Refer to CTS Discussion of Change A03 for Unit 2 and L01 for Unit 1. The successive SR is renumbered to reflect this change.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION (Analog)

B 3.3.13 [Logarithmic] Power Monitoring Channels (Analog)

BASES

BACKGROUND

The [logarithmic] power monitoring channels provide neutron flux power indication from < 1E-7% RTP to > 100% RTP. They also provide reactor protection when the reactor trip circuit breakers (RTCBs) are shut, in the form of a Power Rate of Change - High trip (analog plants) or a [Logarithmic] Power Level - High trip (digital plants).

This LCO addresses MODES 3, 4, and 5 with the RTCBs open. When the RTCBs are shut, the [logarithmic] power monitoring channels are addressed by LCO 3.3.2, "Reactor Protective System (RPS) Instrumentation – Shutdown."

When the RTCBs are open, two of the four wide range power channels must be available to monitor neutron flux power. In this application, the RPS channels need not be OPERABLE since the reactor trip Function is not required. By monitoring neutron flux power when the RTCBs are open, loss of SDM caused by boron dilution can be detected as an increase in flux. Alarms are also provided when power increases above the fixed bistable setpoints. For plants employing separate post accident, [logarithmic] nuclear instrumentation channels with adequate range, these can be substituted for the [logarithmic] power range channels. Two channels must be OPERABLE to provide single failure protection and to facilitate detection of channel failure by providing CHANNEL CHECK capability.

APPLICABLE SAFETY ANALYSES

The [logarithmic] power monitoring channels are necessary to monitor core reactivity changes. They are the primary means for detecting and triggering operator actions to respond to reactivity transients initiated from conditions in which the RPS is not required to be OPERABLE. They also trigger operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions. The [logarithmic] power monitoring channel's LCO requirements support compliance with 10 CFR 50, Appendix A, GDC 13 (Ref. 1). The FSAR, Chapters [7] and [15] (Refs. 2 and 3, respectively), describes the specific [logarithmic] power monitoring channel features that are critical to comply with the GDC.

The OPERABILITY of [logarithmic] power monitoring channels is necessary to meet the assumptions of the safety analyses and provide for the mitigation of accident and transient conditions.

The [logarithmic] power monitoring channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Insert 1 (1)

in MODES 3, 4, and 5. The manual reactor trip Function is required in MODES 3, 4, and 5 with any RTCB closed and any control element assemblies (CEA) capable of being withdrawn. However, by | R1

Insert 2 (2)

The excore wide range logarithmic monitoring channels (i.e., post accident / Appendix R monitors) can be substituted for the wide range logarithmic power range channels. The excore wide range logarithmic monitoring instruments include fission detectors and have a wide range of 2 X 10^{-8}% to 200% RTP and a source range of 1 to 10^{5} cps. The boron dilution alarm channels utilize the excore wide range logarithmic monitoring channels, which subsequently provide an alarm signal to the plant annunciation system upon detection of a dilution event.

BASES

LCO

The LCO on the [logarithmic] power monitoring channels ensures that adequate information is available to verify core reactivity conditions while shut down.

neutron flux

3

A minimum of two [logarithmic] power monitoring channels are required to be OPERABLE. Some plants may have either four or six channels capable of performing this function. In these cases, multiple failures may be tolerated while the plants are still complying with LCO requirements.

neutron flux

3 1

are

APPLICABILITY

In MODES 3, 4, and 5, with RTCBs open or the Control Element Assembly (CEA) Drive System not capable of CEA withdrawal, [logarithmic] power monitoring channels must be OPERABLE to monitor core power for reactivity changes. In MODES 1 and 2, and in MODES 3, 4, and 5 with the RTCBs shut and the CEAs capable of withdrawal, the [logarithmic] power monitoring channels are addressed as part of the RPS in LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating."

neutron flux

wide range

3 1

In MODES 3, 4, and 5 with any RTCB closed and any CEA capable of being withdrawn, manual reactor trip channels are required and addressed by LCO 3.3.2, "Reactor Protection System (RPS) Logic and Trip Initiation."

The requirements for source range neutron flux monitoring in MODE 6 are addressed in LCO 3.9.2, "Nuclear Instrumentation." The source range nuclear instrumentation channels provide neutron flux coverage extending an additional one to two decades below the [logarithmic] channels for use during refueling, when neutron flux may be extremely low. They are built into the [logarithmic] neutron flux channels in the analog plants and in many of the post accident channels used in both the digital and analog plants.

3 1

ACTIONS

A.1 and A.2

With one required channel inoperable, it may not be possible to perform a CHANNEL CHECK to verify that the other required channel is OPERABLE. Therefore, with one or more required channels inoperable, the [logarithmic] power monitoring Function cannot be reliably performed. Consequently, the Required Actions are the same for one required channel inoperable or more than one required channel inoperable. The absence of reliable neutron flux indication makes it difficult to ensure SDM is maintained. Required Action [A.1] is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.

neutron flux

3 1 1 3

BASES

ACTIONS (continued)

SDM must be verified periodically to ensure that it is being maintained. Both required channels must be restored as soon as possible. The initial Completion Time of 4 hours and once every 12 hours thereafter to perform SDM verification takes into consideration that Required Action A.1 eliminates many of the means by which SDM can be reduced. These Completion Times are also based on operating experience in performing the Required Actions and the fact that plant conditions will change slowly.

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1 (1) 2

SR 3.3.13.1 is the performance of a CHANNEL CHECK on each required channel. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based upon the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff and should be based on a combination of the channel instrument uncertainties including control isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

[The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of displays associated with the LCO required channels. 3

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.]

4

SR 3.3.13.2

A CHANNEL FUNCTIONAL TEST is performed to ensure that the entire channel is capable of properly indicating neutron flux. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Internal test circuitry is used to feed preadjusted test signals into the preamplifier to verify channel alignment. It is not necessary to test the detector, because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output. [The Frequency of 92 days is the same as that employed for the same channels in the other applicable MODES.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.]

5

[At this unit, the channel trip Functions tested by the CHANNEL FUNCTIONAL TEST are as follows:]

There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.13.3

1.2

2 5 1

neutron flux

SR 3.3.13.3 is the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is performed. The Surveillance is a complete check and readjustment of the [logarithmic] power channel from the preamplifier input through to the remote indicators. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

1 3

Consistent with the definition of a CHANNEL CALIBRATION, this SR encompasses a CHANNEL FUNCTIONAL TEST.

1

This SR is modified by a Note to indicate that it is not necessary to test the detector because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output. [The Frequency of 18 months is the same as that employed for the same channels in the other applicable MODES.

3

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

4

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
2. FSAR, Chapter [7].
3. FSAR, Chapter [15].

U

1 3

B 3.3 INSTRUMENTATION (Analog)

B 3.3.13 [Logarithmic] Power Monitoring Channels (Analog)

BASES

BACKGROUND

The [logarithmic] power monitoring channels provide neutron flux power indication from < 1E-7% RTP to > 100% RTP. They also provide reactor protection when the reactor trip circuit breakers (RTCBs) are shut, in the form of a Power Rate of Change - High trip (analog plants) or a [Logarithmic] Power Level - High trip (digital plants).

This LCO addresses MODES 3, 4, and 5 with the RTCBs open. When the RTCBs are shut, the [logarithmic] power monitoring channels are addressed by LCO 3.3.2, "Reactor Protective System (RPS) Instrumentation – Shutdown."

When the RTCBs are open, two of the four wide range power channels must be available to monitor neutron flux power. In this application, the RPS channels need not be OPERABLE since the reactor trip Function is not required. By monitoring neutron flux power when the RTCBs are open, loss of SDM caused by boron dilution can be detected as an increase in flux. Alarms are also provided when power increases above the fixed bistable setpoints. For plants employing separate post accident, [logarithmic] nuclear instrumentation channels with adequate range, these can be substituted for the [logarithmic] power range channels. Two channels must be OPERABLE to provide single failure protection and to facilitate detection of channel failure by providing CHANNEL CHECK capability.

APPLICABLE SAFETY ANALYSES

The [logarithmic] power monitoring channels are necessary to monitor core reactivity changes. They are the primary means for detecting and triggering operator actions to respond to reactivity transients initiated from conditions in which the RPS is not required to be OPERABLE. They also trigger operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions. The [logarithmic] power monitoring channel's LCO requirements support compliance with 10 CFR 50, Appendix A, GDC 13 (Ref. 1). The FSAR, Chapters [7] and [15] (Refs. 2 and 3, respectively), describes the specific [logarithmic] power monitoring channel features that are critical to comply with the GDC.

The OPERABILITY of [logarithmic] power monitoring channels is necessary to meet the assumptions of the safety analyses and provide for the mitigation of accident and transient conditions.

The [logarithmic] power monitoring channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Insert 1 (1)

in MODES 3, 4, and 5. The manual reactor trip Function is required in MODES 3, 4, and 5 with any RTCB closed and any control element assemblies (CEA) capable of being withdrawn. However, by | R1

Insert 2 (2)

The excore wide range logarithmic monitoring channels (i.e., post accident / Appendix R monitors) and the startup logarithmic monitoring channels can be substituted for the wide range logarithmic power range channels. The excore wide range logarithmic monitoring instruments include fission detectors and have a wide range of 2 X 10^{-8}% to 200% RTP and a source range of 1 to 10^5 cps. The startup logarithmic monitoring instruments include BF3 detectors and have a range of 1 to 10^5 cps. The boron dilution alarm channels utilize the startup logarithmic monitoring channels, which subsequently provide an alarm signal to the plant annunciation system upon detection of a dilution event.

Neutron Flux

1 3 2 1

BASES

LCO

neutron flux

The LCO on the [logarithmic] power monitoring channels ensures that adequate information is available to verify core reactivity conditions while shut down.

3

neutron flux

Eight

A minimum of two [logarithmic] power monitoring channels are required to be OPERABLE. Some plants may have either four or six channels capable of performing this function. In these cases, multiple failures may be tolerated while the plants are still complying with LCO requirements.

1 3 1

are

APPLICABILITY

In MODES 3, 4, and 5, with RTCBs open or the Control Element Assembly (CEA) Drive System not capable of CEA withdrawal, [logarithmic] power monitoring channels must be OPERABLE to monitor core power for reactivity changes. In MODES 1 and 2, and in MODES 3, 4, and 5 with the RTCBs shut and the CEAs capable of withdrawal, the [logarithmic] power monitoring channels are addressed as part of the RPS in LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating."

neutron flux

3 1

In MODES 3, 4, and 5 with any RTCB closed and any CEA capable of being withdrawn, manual reactor trip channels are required and addressed by LCO 3.3.2, “Reactor Protection System (RPS) Logic and Trip Initiation.”

The requirements for source range neutron flux monitoring in MODE 6 are addressed in LCO 3.9.2, "Nuclear Instrumentation." The source range nuclear instrumentation channels provide neutron flux coverage extending an additional one to two decades below the [logarithmic] channels for use during refueling, when neutron flux may be extremely low. They are built into the [logarithmic] neutron flux channels in the analog plants and in many of the post accident channels used in both the digital and analog plants.

3 1

ACTIONS

A.1 and A.2

With one required channel inoperable, it may not be possible to perform a CHANNEL CHECK to verify that the other required channel is OPERABLE. Therefore, with one or more required channels inoperable, the [logarithmic] power monitoring Function cannot be reliably performed. Consequently, the Required Actions are the same for one required channel inoperable or more than one required channel inoperable. The absence of reliable neutron flux indication makes it difficult to ensure SDM is maintained. Required Action [A.1] is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action, provided they are accounted for in the calculated SDM.

neutron flux

3 1 1 3

St. Lucie Unit 2 1

Revision XXX 2 1

BASES

ACTIONS (continued)

SDM must be verified periodically to ensure that it is being maintained. Both required channels must be restored as soon as possible. The initial Completion Time of 4 hours and once every 12 hours thereafter to perform SDM verification takes into consideration that Required Action A.1 eliminates many of the means by which SDM can be reduced. These Completion Times are also based on operating experience in performing the Required Actions and the fact that plant conditions will change slowly.

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1 (1) 2

SR 3.3.13.1 is the performance of a CHANNEL CHECK on each required channel. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based upon the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff and should be based on a combination of the channel instrument uncertainties including control isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

[The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of displays associated with the LCO required channels. 3

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------] 4

SR 3.3.13.2

A CHANNEL FUNCTIONAL TEST is performed to ensure that the entire channel is capable of properly indicating neutron flux. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Internal test circuitry is used to feed preadjusted test signals into the preamplifier to verify channel alignment. It is not necessary to test the detector, because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output. [The Frequency of 92 days is the same as that employed for the same channels in the other applicable MODES.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------] 5

[At this unit, the channel trip Functions tested by the CHANNEL FUNCTIONAL TEST are as follows:]

There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.13.3

1.2

2 5 1

neutron flux

SR 3.3.13.3 is the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is performed. The Surveillance is a complete check and readjustment of the [logarithmic] power channel from the preamplifier input through to the remote indicators. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillances. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

1 3

Consistent with the definition of a CHANNEL CALIBRATION, this SR encompasses a CHANNEL FUNCTIONAL TEST.

1

This SR is modified by a Note to indicate that it is not necessary to test the detector because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output. [The Frequency of 18 months is the same as that employed for the same channels in the other applicable MODES.

3

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

4

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
2. FSAR, Chapter [7].
3. FSAR, Chapter [15].

U

1 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.11 BASES, LOGARITHMIC NEUTRON FLUX MONITORING

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Changes are made to reflect the ITS Specification number and title. Numbering is changed from 3.3.13 to 3.3.11 and "(Analog)" is deleted from the title.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. ISTS SR 3.3.13.2 Bases discussion is not included in the ITS to reflect the changes to the Specification in the ITS. The successive SR is renumbered to reflect this change.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.11, LOGARITHMIC NEUTRON FLUX MONITORING

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 12

ISTS Not Adopted

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.3 INSTRUMENTATION (Analog)

3.3.2 Reactor Protective System (RPS) Instrumentation - Shutdown (Analog)

LCO 3.3.2 Four Power Rate of Change - High RPS trip units and associated instrument and bypass removal channels shall be OPERABLE.

APPLICABILITY: MODES 3, 4, and 5, with any reactor trip circuit breakers (RTCBs) closed and any control element assembly capable of being withdrawn.

NOTE

Trip may be bypassed when THERMAL POWER is < [1E-4]% RTP. Bypass shall be automatically removed when THERMAL POWER is ≥ [1E-4]% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Power Rate of Change - High trip unit or associated instrument channel inoperable.	A.1 Place affected trip unit in bypass or trip.	1 hour
	AND	
	A.2.1 Restore channel to OPERABLE status.	[48] hours
	OR	
	A.2.2 [Place affected trip unit in trip.	48 hours]

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two Power Rate of Change - High trip units or associated instrument channel inoperable.	B.1 Place one trip unit in bypass and place the other trip unit in trip. AND B.2 [Restore one trip unit to OPERABLE status.	1 hour 48 hours]
C. One automatic bypass removal channel inoperable.	C.1 Disable bypass channel. OR C.2.1 Place affected trip unit in bypass or trip. AND C.2.2.1 Restore bypass removal channel and affected trip unit to OPERABLE status. OR C.2.2.2 [Place affected trip units in trip.	1 hour 1 hour [48] hours 48 hours]
D. Two automatic bypass removal channels inoperable.	D.1 Disable bypass channels. OR D.2.1 Place one affected trip unit in bypass and place the other in trip. AND	1 hour 1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	D.2.2 Restore one bypass channel and the associated trip unit to OPERABLE status.	[48] hours
E. Required Action and associated Completion Time not met.	E.1 Open all RTCBs.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform a CHANNEL CHECK of each wide range power channel.	[12 hours OR In accordance with the Surveillance Frequency Control Program]
SR 3.3.2.2	Perform a CHANNEL FUNCTIONAL TEST on the Power Rate of Change trip function.	[92 days OR In accordance with the Surveillance Frequency Control Program]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.3	Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal function.	[92 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.3.2.4	NOTE Neutron detectors are excluded from CHANNEL CALIBRATION. Perform a CHANNEL CALIBRATION, including bypass removal functions with Allowable Value ≤ [2.6] dpm.	[[18] months OR In accordance with the Surveillance Frequency Control Program]

JUSTIFICATION FOR DEVIATIONS
ISTS 3.3.2, REACTOR PROTECTIVE SYSTEM (RPS) INSTRUMENTATION - SHUTDOWN (ANALOG)

1. ISTS 3.3.2, "Reactor Protective System (RPS) Instrumentation - Shutdown (Analog)," Specification is not included in the St. Lucie Plant (PSL) Unit 1 and Unit 2 ITS because the RPS Rate of Change trip function is not credited in any UFSAR Chapter 15 accident analysis. Monitoring of the function is addressed by ITS 3.3.11, “Logarithmic Neutron Flux Monitoring.”

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

1

B 3.3 INSTRUMENTATION (Analog)

B 3.3.2 Reactor Protective System (RPS) Instrumentation - Shutdown (Analog)

BASES

BACKGROUND The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and reactor coolant pressure boundary integrity during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also assists the Engineered Safety Features systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling,
- Fuel centerline melting shall not occur, and
- The Reactor Coolant System pressure SL of [2750] psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 1) and 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

1

BASES

BACKGROUND (continued)

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels,
- Bistable trip units,
- RPS Logic, and
- Reactor trip circuit breakers (RTCBs).

This LCO applies only to the Power Rate of Change - High trip Functions and associated instrument channels in MODES 3, 4, and 5 with any of the RTCBs closed and any Control Element Assembly (CEA) capable of being withdrawn. In MODES 1 and 2, this trip Function is addressed in LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating." LCO 3.3.13, "[Logarithmic] Power Monitoring Channels," applies when the RTCBs are open or CEA Drive System is not capable of CEA withdrawal. In the case of LCO 3.3.13, the logarithmic power instrumentation channels are required for monitoring neutron flux, although the trip Function is not required.

Measurement Channels and Trip Units

The measurement channels providing input to the Power Rate of Change - High Function consist of wide range nuclear instrumentation channels using neutron flux leakage from the reactor vessel.

Other aspects of the Power Rate of Change - High trip are similar to the other measurement channels and bistable trip units. These are addressed in the Background section of LCO 3.3.1.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Reference 3 takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. Other Functions, such as the Loss of Load trip, are purely equipment protective, and their use minimizes the potential for equipment damage.

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Power Rate of Change - High trip is used to trip the reactor when excore wide range power indicates an excessive rate of change.

The Power Rate of Change - High trip is not required for protection. It serves as a backup to the administratively enforced startup rate limit.

The Power Rate of Change - High Function minimizes transients for events such as a continuous CEA withdrawal or a boron dilution event from low power levels. The Power Rate of Change - High trip is automatically bypassed at < 1E-4% RTP, as sensed by the wide range nuclear instrument (NI) Level 2 bistable, when poor counting statistics may lead to erroneous indication. It is also bypassed at > 12% RTP, where moderator temperature coefficient and fuel temperature coefficient make high rate of change of power unlikely. This bypass is effected by the power range NI Level 1 bistable. Automatic bypass removal is also effected by these bistables. With the RTCBs open, the Power Rate of Change - High trip is not required to be OPERABLE; however, the indication and alarm Functions of at least two channels are required to be OPERABLE. LCO 3.3.13 ensures the wide range channels are available to detect and alert the operator to a boron dilution event.

The RPS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any required portion of the instrument channel or bypass removal channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

Actions allow maintenance (trip channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel in the same Function bypassed. Plants are restricted to 48 hours in a trip channel bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic). At plants where adequate channel to channel independence has been demonstrated, specific exceptions have been approved by the NRC staff to permit one of the two-out-of-four channels to be bypassed for an extended period of time.

This LCO requires four channels of Power Rate of Change - High to be OPERABLE in MODES 3, 4, and 5, when the RTCBs are closed and the CEA Drive System is capable of CEA withdrawal. MODE 1 and 2 requirements are addressed in LCO 3.3.1. This trip is not credited in the safety analysis. Therefore, the Allowable Value specified in SR 3.3.4.2 is not derived from an analytical limit.

1

BASES

APPLICABILITY This LCO is applicable to the Power Rate of Change - High reactor trip in MODES 3, 4 and 5. MODES 1 and 2 are addressed in LCO 3.3.1.

The power rate of change trip is required in MODES 3, 4, and 5, with the RTCBs closed and a CEA capable of being withdrawn to provide backup protection for boron dilution and CEA withdrawal events. The power rate of change trip is not credited in the safety analysis, but is part of the NRC approved licensing basis for the plant.

The power rate of change trip has operating bypasses discussed in the LCO section. In MODES 3, 4, and 5, the emphasis is placed on return to power events. The reactor is protected in these MODES by ensuring adequate SDM.

ACTIONS The most common causes of channel inoperability are outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the trip setpoint is less conservative than the Allowable Value in Table 3.3.1-1, the channel is declared inoperable immediately, and the appropriate Condition(s) must be entered immediately.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or RPS bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the plant must enter the Condition for the particular protection Function affected.

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

A.1, A.2.1, and A.2.2

Condition A applies to the failure of a single channel of the Power Rate of Change - High RPS automatic trip Function.

BASES

ACTIONS (continued)

RPS coincidence logic is normally two-out-of-four. If one RPS bistable trip unit or associated instrument channel is inoperable, startup or power operation is allowed to continue, providing the inoperable trip unit is placed in bypass or trip within 1 hour (Required Action A.1). With one channel in bypass, no additional random failure of a single channel could spuriously trip the reactor and a valid trip signal can still trip the reactor. With one channel in trip, an additional random failure of a single channel could spuriously trip the reactor. Therefore, it is preferable to place an inoperable channel in bypass rather than trip.

The Completion Time of 1 hour allotted to restore, bypass, or trip the channel is sufficient to allow the operator to take all appropriate actions for the failed channel, while ensuring that the risk involved in operating with the failed channel is acceptable.

For plants that have not demonstrated sufficient channel to channel independence, the failed channel is restored to OPERABLE status or is placed in trip within 48 hours (Required Action A.2.1 or Required Action A.2.2). Required Action A.2.1 restores the full capability of the Function. Required Action A.2.2 places the Function in a one-out-of-three configuration. In this configuration, common cause failure of dependent channels cannot prevent trip.

The [48] hour Completion Time is based on operating experience, which has demonstrated that a random failure of a second channel occurring during the [48] hour period is a low probability event.

B.1 and B.2

Condition B applies to the failure of two channels in the Power Rate of Change - High RPS automatic trip Function.

Required Action B.1 provides for placing one inoperable channel in bypass and the other channel in trip within the Completion Time of 1 hour. This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels, while ensuring the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed, the RPS is in a two-out-of-three logic; but with another channel failed, the RPS may be operating in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

BASES

ACTIONS (continued)

[The bypassed channel should be restored to OPERABLE status within 48 hours for reasons similar to those stated under Condition A. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 shall be placed in trip if more than 48 hours have elapsed since the initial channel failure.]

C.1, C.2.1, C.2.2.1, and C.2.2.2

Condition C applies to one automatic bypass removal channel inoperable. If the bypass removal channel cannot be restored to OPERABLE status, the associated Power Rate of Change - High RPS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected RPS channel must be declared inoperable, as in Condition A, and the bypass either removed or the bypass removal channel repaired. The Bases for the Required Actions and Completion Times are the same as discussed for Condition A.

D.1, D.2.1, and D.2.2

Condition D applies to two inoperable automatic bypass removal channels. If the bypass removal channels cannot be restored to OPERABLE status, the associated Power Rate of Change - High RPS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected RPS channels must be declared inoperable, as in Condition B, and the bypass either removed or the bypass removal channel repaired. Also, Required Action D.2.2 provides for the restoration of the one affected automatic trip channel to OPERABLE status within the rules of Completion Time specified under Condition B. Completion Times are consistent with Condition B.

E.1

Condition E is entered when the Required Actions and associated Completion Times of Condition A, B, C, or D are not met.

1

BASES

ACTIONS (continued)

If Required Actions associated with these Conditions cannot be completed within the required Completion Time, opening the RTCBs brings the reactor to a MODE where the LCO does not apply and ensures no CEA withdrawal will occur. The basis for the Completion Time of 6 hours is that it is adequate to complete the Required Actions without challenging plant systems, including the insertion of CEAs for plants that normally maintain CEAs withdrawn when shut down.

SURVEILLANCE REQUIREMENTS

----------REVIEWER'S NOTE----------
In order for a plant to take credit for topical reports as the basis for justifying Frequencies, topical reports must be supported by an NRC staff Safety Evaluation Report that establishes the acceptability of each topical report for that plant.

SR 3.3.2.1

Performance of the CHANNEL CHECK on each wide range channel ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on another channel. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

[The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

SR 3.3.2.2

A CHANNEL FUNCTIONAL TEST on the power rate of change channels is performed to ensure the entire channel will perform its intended function if required. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Power Rate of Change - High trip Function is required during startup operation and is bypassed when shut down or > 15% RTP. [Additionally, operating experience has shown that these components usually pass the Surveillance when performed at a Frequency of once every 92 days prior to each reactor startup. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.3

SR 3.3.2.3 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.2.2, except SR 3.3.2.3 is applicable only to bypass Functions and is performed prior to each startup. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Proper operation of bypass permissives is critical during plant startup because the bypasses must be in place to allow startup operation and must be removed at the appropriate points during power ascent to enable certain reactor trips. Consequently, the appropriate time to verify bypass removal function OPERABILITY is just prior to startup. [The allowance to conduct this Surveillance within 92 days of startup is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 5).

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

Once the operating bypasses are removed, the bypasses must not fail in such a way that the associated trip Function gets inadvertently bypassed. This feature is verified by the trip Function CHANNEL FUNCTIONAL TEST, SR 3.3.2.2. Therefore, further testing of the bypass function after startup is unnecessary.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.4

SR 3.3.2.4 is the performance of a CHANNEL CALIBRATION.

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

Only the Allowable Values are specified for each RPS trip Function. Nominal trip setpoints are specified in the plant specific setpoint calculations. The nominal setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip Function. These uncertainties are defined in the "Plant Protection System Selection of Trip Setpoint Values" (Ref. 4).

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in Reference 5.

[The Frequency is based upon the assumption of an [18] month calibration interval in the determination of the magnitude of equipment drift.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

The Surveillance is modified by a Note to indicate that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal.

REFERENCES

1. 10 CFR 50, Appendix A.

2. 10 CFR 100.

3. FSAR, Chapter [14].

4. "Plant Protection System Selection of Trip Setpoint Values."

5. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989.

JUSTIFICATION FOR DEVIATIONS
ISTS 3.3.2 BASES, REACTOR PROTECTIVE SYSTEM (RPS) INSTRUMENTATION - SHUTDOWN (ANALOG)

1. ISTS 3.3.2, "Reactor Protective System (RPS) Instrumentation - Shutdown (Analog)," Specification is not included in the St. Lucie Plant (PSL) Unit 1 and Unit 2 ITS because the RPS Rate of Change function is not credited in any UFSAR Chapter 15 accident analysis. Monitoring of the function is addressed by ITS 3.3.11, “Logarithmic Neutron Flux Monitoring.” Therefore, the Bases associated with ISTS 3.3.2 is not included in the PSL Unit 1 and Unit 2 ITS Bases.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

1

3.3 INSTRUMENTATION (Analog)

3.3.9 Chemical and Volume Control System (CVCS) Isolation Signal (Analog)

LCO 3.3.9 Four channels of West Penetration Room/Letdown Heat Exchanger Room pressure sensing and two Actuation Logic channels shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Actuation Logic channel inoperable.	A.1 Restore the channel to OPERABLE status.	48 hours
B. One CVCS isolation instrument channel inoperable.	B.1 Place the channel in bypass or trip. AND B.2.1 Restore the channel to OPERABLE status. OR B.2.2 Place the channel in trip.	1 hour 48 hours 48 hours
C. Two CVCS isolation instrument channels inoperable.	C.1 Place one channel in bypass and place the other channel in trip. AND C.2 Restore one channel to OPERABLE status.	1 hour 48 hours

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours
E. Two Actuation Logic channels inoperable.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.9.1 Perform a CHANNEL CHECK.	[12 hours OR In accordance with the Surveillance Frequency Control Program]

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.9.2 NOTES 1. Testing of Actuation Logic shall include the verification of the proper operation of each initiation relay. 2. Relays associated with plant equipment that cannot be operated during plant operation are only required to be tested during each MODE 5 entry exceeding 24 hours unless tested within the previous 6 months. Perform a CHANNEL FUNCTIONAL TEST on each CVCS isolation channel with setpoints in accordance with the following Allowable Values: West Penetration Room Pressure - High ≤ .5 psig Letdown Heat Exchanger Room Pressure - High ≤ .5 psig	[31 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.3.9.3 Perform a CHANNEL CALIBRATION on each CVCS isolation pressure indicating channel.	[18 months OR In accordance with the Surveillance Frequency Control Program]

JUSTIFICATION FOR DEVIATIONS
ISTS 3.3.9, CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)
ISOLATION SIGNAL

1. ISTS 3.3.9, "Chemical and Volume Control System (CVCS) Isolation Signal, "Specification is not included in the St. Lucie Plant (PSL) Unit 1 and Unit 2 ITS because PSL Unit 1 and Unit 2 does not include a Chemical and Volume Control System Isolation Signal.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

1

B 3.3 INSTRUMENTATION (Analog)

B 3.3.9 Chemical and Volume Control System (CVCS) Isolation Signal (Analog)

BASES

BACKGROUND This LCO encompasses Chemical and Volume Control System (CVCS) Isolation Signal actuation. This is a plant specific instrumentation channel that performs an actuation Function required for plant protection and is not otherwise included in LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip," or LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS)." This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements, is not included in LCOs 3.3.5 and 3.3.6. Details of this LCO are for illustration only. Individual plants shall include those Functions and LCO requirements that are applicable to them.

The CVCS Isolation Signal provides protection from radioactive contamination, as well as personnel and equipment protection in the event of a letdown line rupture outside containment.

Each of the two actuation subsystems will isolate a separate letdown isolation valve in response to a high pressure condition in either the West Penetration Room or Letdown Heat Exchanger Room. Two pressure detectors in each of these rooms feed the four sensor subsystems. On a two-out-of-four coincidence, both actuation subsystems will actuate.

Trip Setpoints and Allowable Values

Trip setpoints used in the bistables are based on the analytical limits stated in Reference 1. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Values specified in SR 3.3.9.2 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in "Plant Protection System Selection of Trip Setpoint Values" (Ref. 2). The actual nominal trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

1

BASES

BACKGROUND (continued)

Setpoints in accordance with the Allowable Value will ensure that Safety Limits are not violated during anticipated operational occurrences (AOOs) and the consequences of Design Basis Accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or accident and the equipment functions as designed.

APPLICABLE SAFETY ANALYSES

The CVCS Isolation Signal is redundant to the Safety Injection Actuation Signal for letdown line breaks outside containment. In addition, an excess flow check valve is located in containment just downstream of the regenerative heat exchanger, which isolates letdown when flow exceeds 200 gpm.

[At this unit, the provision of two sensors in each room in a two-out-of-four logic configuration satisfies the single failure criterion as follows:]

The CVCS satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Only the Allowable Values are specified for each trip Function in the LCO. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that the difference between the nominal trip setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the transient and accident analyses.

Each Allowable Value specified is more conservative than the analytical limit assumed in the transient and accident analysis, in order to account for instrument uncertainties appropriate to the trip Function. These uncertainties are defined in the "Plant Protection System Selection of Trip Setpoint Values" (Ref. 2).

CVCS isolation consists of closing the appropriate valve. This is undesirable at power, since letdown isolation will result. The absence of letdown flow will significantly decrease the charging flow temperature due to the absence of the regenerative heat exchanger preheating, causing unnecessary thermal stress to the charging nozzle. Therefore, the preferred action is to restore the valve function to OPERABLE status.

Four channels of West Penetration Room and Letdown Heat Exchanger Room pressure sensing and two Actuation Logic channels are required to be OPERABLE.

[For this unit, the Bases for the Allowable Values are as follows:]

[For this unit, the Bases for the LCO requirement are as follows:]

1

BASES

APPLICABILITY The CVCS Isolation Signal must be OPERABLE in MODES 1, 2, 3, and 4, since the possibility of a loss of coolant accident is greatest in these MODES. In MODE 5 or 6, the probability is greatly diminished, and there is time to manually isolate CVCS.

ACTIONS A CVCS isolation channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function. The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific setpoint analysis. Typically, the drift is not large and would result in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CHANNEL FUNCTIONAL TEST, when the process instrument is set up for adjustment to bring it within specification. If the trip setpoint is not consistent with the Allowable Value in SR 3.3.9.2, the channel must be declared inoperable immediately and the appropriate Conditions must be entered.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the sensor, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel should be declared inoperable and the LCO Condition entered for the particular protection Function affected.

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to the failure of one CVCS Actuation Logic channel associated with the CVCS Isolation Signal. Required Action A.1 requires restoration of the inoperable channel to restore redundancy of the affected Function. The Completion Time of 48 hours is consistent with the Completion Time of other ESFAS Functions and should be adequate for most repairs, while minimizing the risk of operating with an inoperable channel.

1

BASES

ACTIONS (continued)

B.1, B.2.1, and B.2.2

Condition B applies if one of the four CVCS instrument channels is inoperable. The Required Actions are identical to those of ESFAS Functions employing four redundant sensors specified in LCO 3.3.4, "Engineered Safety Features Actuation System (ESFAS) Instrumentation." The channel must be placed in bypass or trip if it cannot be repaired within 1 hour (Required Action B.1). The provision of four trip channels allows one channel to be bypassed (removed from service) during operations, placing the ESFAS in two-out-of-three coincidence logic. Placing the channel in bypass is preferred, since the CVCS isolation Function will be in two-out-of-three logic. This will avoid possible inadvertent CVCS isolation if an additional channel fails. The 1 hour Completion Time to bypass or trip the channel is sufficient time to perform the Required Actions.

Once the Required Action to trip or bypass the channel has been complied with, Required Action B.2.1 and Required Action B.2.2 provide for restoring the channel to OPERABLE status or placing it in trip within 48 hours. Required Action B.2.1 restores the full capability of the Function. Required Action B.2.2 places the Function in a one-out-of-three configuration. In this configuration, common cause failure of dependent channels cannot prevent CVCS isolation actuation. The Completion Time provides the operator with time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. It is improbable that a failure of a second channel will occur during any given 48 hour period.

C.1 and C.2

Condition C applies if two of the four CVCS West Penetration Room/Letdown Heat Exchanger Room Pressure - High channels are inoperable. The Required Actions are identical to those for other ESFAS Functions employing four redundant sensors in LCO 3.3.4.

Restoring at least one channel to OPERABLE status is the preferred Required Action. If this cannot be accomplished, one channel should be placed in bypass and the other channel in trip. The allowed Completion Time of 1 hour is sufficient time to perform the Required Actions.

Once the Required Action to trip or bypass the channel has been complied with, Required Action C.2 provides for restoring one channel to OPERABLE status within 48 hours. The justification of the 48 hour Completion Time is the same as for Condition B.

1

BASES

ACTIONS (continued)

After one channel is restored to OPERABLE status, the provisions of Condition C still apply to the remaining inoperable channel.

D.1 and D.2

---------- REVIEWER'S NOTE ----------
Adoption of a MODE 4 end state requires the licensee to make the following commitments:

1. [Licensee] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].

2. [Licensee] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.

Condition D is entered when the Required Actions and associated Completion Times of Condition A, B, or C are not met. The plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 3). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Required Action D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is

1

BASES

ACTIONS (continued)

not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

E.1 and E.2

Condition E specifies the shutdown track to be followed if two Actuation Logic channels are inoperable. The plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK on each CVCS isolation pressure indicating channel ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

[The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

SR 3.3.9.2

A CHANNEL FUNCTIONAL TEST is performed on each channel to ensure the entire channel will perform its intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint analysis. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

[The Frequency of 31 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is a rare event.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

Proper operation of the individual subgroup relays is verified by de-energizing these relays during the CHANNEL FUNCTIONAL TEST of the Actuation Logic. This will actuate the Function, operating all associated equipment. Proper operation of the equipment actuated by each train is thus verified. Note 1 indicates this test includes verification of operation for each initiation relay. [At this unit, the verification is conducted as follows:]

Note 2 indicates that relays that cannot be tested at power are excepted from the SR while at power. These relays must, however, be tested during each entry into MODE 5 exceeding 24 hours unless they have been tested within the previous 6 months.

[At this unit, the basis for this test exception is as follows:]

[At this unit, the following relays excepted by this Note are:]

SR 3.3.9.3

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. There is a plant specific program which verifies that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. CHANNEL CALIBRATIONS must be performed consistent with the plant specific setpoint analysis.

Radiation detectors may be removed and calibrated in a laboratory, calibrated in place using a transfer source or replaced with an equivalent laboratory calibrated unit.

[The Frequency is based upon the assumptions of an 18 month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis and includes operating experience as well as consistency with an 18 month fuel cycle.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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REFERENCES

1. FSAR, Section [7.3].

2. "Plant Protection System Selection of Trip Setpoint Values."

3. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

JUSTIFICATION FOR DEVIATIONS
ISTS 3.3.9 BASES, CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) ISOLATION SIGNAL

1. ISTS 3.3.9, "Chemical and Volume Control System (CVCS) Isolation Signal, "Specification is not included in the St. Lucie Plant (PSL) Unit 1 and Unit 2 ITS because PSL Unit 1 and Unit 2 does not include a Chemical and Volume Control System Isolation Signal. Therefore, the Bases associated with ISTS 3.3.9 is not included in the PSL Unit 1 and Unit 2 ITS Bases.

ENCLOSURE 2

VOLUME 9

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.4 REACTOR COOLANT SYSTEM (RCS)

Revision 1

LIST OF ATTACHMENTS

1. 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
2. 3.4.2, RCS Minimum Temperature for Criticality
3. 3.4.3, RCS Pressure and Temperature (P/T) Limits
4. 3.4.4, RCS Loops - MODES 1 and 2
5. 3.4.5, RCS Loops - MODE 3
6. 3.4.6, RCS Loops - MODE 4
7. 3.4.7, RCS Loops - MODE 5, Loops Filled
8. 3.4.8, RCS Loops - MODE 5, Loops Not Filled
9. 3.4.9, Pressurizer
10. 3.4.10, Pressurizer Safety Valves
11. 3.4.11, Pressurizer Power Operated Relief Valves (PORVs)
12. 3.4.12, Low Temperature Overpressure Protection (LTOP) System
13. 3.4.13, RCS Operational LEAKAGE
14. 3.4.14, RCS Pressure Isolation Valve (PIV) Leakage
15. 3.4.15, RCS Leakage Detection Instrumentation
16. 3.4.16, RCS Specific Activity
17. 3.4.17, Steam Generator (SG) Tube Integrity
18. Relocated/Deleted Current Technical Specifications
19. ISTS Not Adopted

ATTACHMENT 1

3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

~~POWER DISTRIBUTION LIMITS~~ ← 3.4 REACTOR COOLANT SYSTEM (RCS)

~~DNB PARAMETERS~~ ← 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LIMITING CONDITION FOR OPERATION

LCO 3.4.1 ~~3.2.5~~ ~~The following~~ RCS DNB ~~related~~ parameters shall be ~~maintained~~ within the limits:

SR 3.4.1.2 ~~a.~~ Cold Leg Temperature ~~as shown on Table 3.2-1 of the COLR,~~

SR 3.4.1.1 ~~b.~~ Pressurizer Pressure* ~~as shown on Table 3.2-1 of the COLR,~~ ← specified in the COLR

SR 3.4.1.3 ~~c.~~ Reactor Coolant System Total Flow Rate ~~greater than or equal to~~ ≥ 375,000 gpm~~, and~~

d. AXIAL SHAPE INDEX as shown on Figure 3.2-4 of the COLR. — See ITS 3.2.4

Applicability **APPLICABILITY**: MODE 1.

ACTION:

ACTION A With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or ~~reduce THERMAL POWER to < 5% of RATED THERMAL POWER~~ within

ACTION B ~~the next~~ 4 hours. ← 6 ← Be in MODE 2

L02 R1

SURVEILLANCE REQUIREMENTS

Pressurizer pressure
RCS cold leg temperature
RCS total flow rate

SR 3.4.1.1
SR 3.4.1.2
SR 3.4.1.3 ~~4.2.5.1~~ Each of the DNB ~~related~~ parameters shall be verified to be within their limits ~~by instrument readout~~ in accordance with the Surveillance Frequency Control Program.

SR 3.4.1.4 ~~4.2.5.2~~ ~~The Reactor Coolant System~~ RCS total flow rate shall be determined to be within its limit by ~~measurement~~** in accordance with the Surveillance Frequency Control Program. ← precision heat balance

SR 3.4.1.4 Note

Pressurizer pressure

Applicability Note * Limit not applicable during either a THERMAL POWER ramp ~~increase in excess of~~ > 5% RTP per minute ~~of RATED THERMAL POWER~~ or a THERMAL POWER step ~~increase of greater than~~ > 10% of RATED THERMAL POWER.

SR 3.4.1.4 Note ** Not required to be performed until THERMAL POWER is ≥ 90% of RATED THERMAL POWER. ← 24 hours after

L01

A01

~~POWER DISTRIBUTION LIMITS~~ ← 3.4 REACTOR COOLANT SYSTEM (RCS)

~~DNB PARAMETERS~~ ← 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LIMITING CONDITION FOR OPERATION

LCO 3.4.1 ~~3.2.5~~ ~~The following~~ RCS DNB ~~related~~ parameters shall be ~~maintained~~ within the limits:

~~a.~~ Cold Leg Temperature ~~as shown on Table 3.2-2 of the COLR,~~

~~b.~~ Pressurizer Pressure* ~~as shown on Table 3.2-2 of the COLR,~~ ← specified in the COLR

~~c.~~ Reactor Coolant System Total Flow Rate ~~greater than or equal to~~ ≥ 375,000 gpm~~, and~~

d. AXIAL SHAPE INDEX as shown on Figure 3.2-4 of the COLR. — See ITS 3.2.4

Applicability **APPLICABILITY**: MODE 1.

ACTION:

ACTION A With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or ~~reduce THERMAL POWER to < 5% of RATED THERMAL POWER~~ within ~~the next~~ ~~4~~ 6 hours. L02 R1

ACTION B ← Be in MODE 2

Pressurizer pressure
RCS cold leg temperature
RCS total flow rate

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1
SR 3.4.1.2
SR 3.4.1.3 ~~4.2.5.1~~ Each of the DNB ~~related~~ parameters shall be verified to be within their limits ~~by instrument readout~~ in accordance with the Surveillance Frequency Control Program.

SR 3.4.1.4 ~~4.2.5.2~~ ~~The Reactor~~ RCS ~~Coolant System~~ total flow rate shall be determined to be within its limit by ~~measurement~~** in accordance with the Surveillance Frequency Control Program.

precision heat balance

Pressurizer pressure

Applicability Note * Limit not applicable during either a THERMAL POWER ramp ~~increase in excess of~~ > 5% RTP per minute ~~of RATED THERMAL POWER~~ or a THERMAL POWER step ~~increase of greater than~~ > 10% of RATED THERMAL POWER.

SR 3.4.1.4 Note ** Not required to be performed until 24 hours after THERMAL POWER is ≥ 90% of RATED THERMAL POWER. L01

DISCUSSION OF CHANGES
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.2.5.2 Note states "Not required to be performed until THERMAL POWER is ≥ 90% of RATED THERMAL POWER. ITS SR 3.4.1.4 states "Not required to be performed until 24 hours after ≥ 90% RTP. ITS SR 3.4.1.4 Note allows 24 hours after ≥ 90% RTP to perform the Surveillance. This is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance will not yield accurate results if performed below 90% RTP. ITS SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Once the unit reaches 90% RTP, a 24 hour allowance is provided for completing the Surveillance. If the Surveillance were not performed within this 24 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply. This changes the CTS by allowing up to 24 hours after ≥ 90% RTP to perform the Surveillance.

The purpose of CTS 4.2.5.2 is to perform the Surveillance at normal operating conditions when ≥ 90% RTP. ITS SR 3.4.1.4 allows up to 24 hours after ≥ 90% RTP to perform the Surveillance. This change is acceptable because the "24 hours" allows a reasonable time to perform the Surveillance once establishing

the conditions at which the Surveillances can be performed. Additionally, the proposed time to perform the Surveillance is acceptable because there are other plant parameters available to indicate significant RCS low flow conditions. For example; reactor coolant pump running status indication, RCS low flow alarm, and Reactor Coolant Flow – low reactor trip. This change is designated as less restrictive because the Surveillances will be performed within an allowance of 24 hours after ≥ 90% RTP rather than immediately after ≥ 90% RTP.

L02 (*Category 3 – Relaxation of Completion Time*) CTS 3.2.5 Action requires the unit to reduce THERMAL POWER to ≤ 5% of RTP within the next 4 hours if the DNB parameters are not restored to within limit in 2 hours. ITS 3.4.1 ACTION B requires the power reduction to ≤ 5% RTP (MODE 2) within the next 6 hours if the DNB parameters are not restored to within limit in 2 hours. This changes the CTS by extending the time for the unit to be placed outside the MODE of Applicability. R1

The purpose of the CTS 3.2.5 Action is to limit the time the unit can be outside of the DNB parameter limits and remain within the Applicability of the Specification. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the low probability of a DBA occurring during the allowed Completion Time. The change extends the time the unit is allowed to be outside the DNB parameter limits and be in the Applicability of the Specification. The time extension from 4 hours to 6 hours is consistent with the ISTS Completion Time to place the unit in MODE 2 and is considered a reasonable time that permits the plant power to be reduced at an orderly rate in conjunction with even control of steam generator heat removal. This change is designated as less restrictive because additional time is allowed to be outside the DNB parameter limits and be in the Applicability of the Specification than was allowed in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow [Departure from Nucleate Boiling (DNB)] Limits

2

LCO 3.2.5.a
LCO 3.2.5.b
LCO 3.2.5.c

LCO 3.4.1 RCS DNB parameters for pressurizer pressure[,] and cold leg temperature, and RCS total flow rate shall be within the limits specified in the COLR.

and

shall be ≥ 375,000 gpm

4

3.2.5 Applicability

APPLICABILITY: MODE 1.

----------NOTE----------
LCO 3.2.5.b Footnote *

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.2.5 Action A. [Pressurizer pressure or RCS flow rate] not within limits. RCS DNB parameter(s)	A.1 Restore parameter(s) to within limit.	2 hours
3.2.5 Action B. Required Action and associated Completion Time [of Condition A] not met.	B.1 Be in MODE 2.	6 hours
[C. RCS cold leg temperature not within limits.]	[C.1 Restore cold leg temperature to within limits.]	[2 hours]
[D. Required Action and associated Completion Time of Condition C not met.]	[D.1 Reduce THERMAL POWER to ≤ [30]% RTP.]	[6 hours]

3

3

3

1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.2.5.1 LCO 3.2.5.b	SR 3.4.1.1	Verify pressurizer pressure is within the limits specified in the COLR.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.2.5.1 LCO 3.2.5.a	SR 3.4.1.2	Verify RCS cold leg temperature is within the limits specified in the COLR.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.2.5.1 LCO 3.2.5.c	SR 3.4.1.3	~~NOTE~~ ~~Only required to be met in MODE 1.~~ Verify RCS total flow rate is ~~greater than or equal to the limits specified in the COLR.~~ ≥ 375,000 gpm	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	R1 5 4 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.1.4	----------NOTE---------- Not required to be performed until ~~[~~24~~]~~ hours after ≥ ~~[~~90~~]~~% RTP. ---------- Verify by precision heat balance that RCS total flow rate is ~~within the limits specified in the COLR~~. ≥ 375,000 gpm	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

4.2.5.2 Footnote ** LCO 3.2.5.c

DOC L01

R1

4 2

2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow [Departure from Nucleate Boiling (DNB)] Limits

2

LCO 3.2.5.a
LCO 3.2.5.b
LCO 3.2.5.c

LCO 3.4.1 RCS DNB parameters for pressurizer pressure[,] and cold leg temperature, and RCS total flow rate shall be within the limits specified in the COLR.

and

shall be ≥ 375,000 gpm

4

3.2.5 Applicability

APPLICABILITY: MODE 1.

----------NOTE----------

LCO 3.2.5.b Footnote *

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.2.5 Action A. [Pressurizer pressure or RCS flow rate] not within limits. RCS DNB parameter(s)	A.1 Restore parameter(s) to within limit.	2 hours
3.2.5 Action B. Required Action and associated Completion Time [of Condition A] not met.	B.1 Be in MODE 2.	6 hours
[C. RCS cold leg temperature not within limits.]	[C.1 Restore cold leg temperature to within limits.]	[2 hours]
[D. Required Action and associated Completion Time of Condition C not met.]	[D.1 Reduce THERMAL POWER to ≤ [30]% RTP.]	[6 hours]

3

3

3

1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.2.5.1 LCO 3.2.5.b	SR 3.4.1.1	Verify pressurizer pressure is within the limits specified in the COLR.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.2.5.1 LCO 3.2.5.a	SR 3.4.1.2	Verify RCS cold leg temperature is within the limits specified in the COLR.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.2.5.1 LCO 3.2.5.c	SR 3.4.1.3	~~NOTE~~ ~~Only required to be met in MODE 1.~~ Verify RCS total flow rate is ~~greater than or equal to the limits specified in the COLR.~~ ≥ 375,000 gpm	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	R1 5 4 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.1.4	----------NOTE---------- Not required to be performed until [24] hours after ≥ [90]% RTP. ----------	
	Verify by precision heat balance that RCS total flow rate is ~~within the limits specified in the COLR~~. ≥ 375,000 gpm	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

4.2.5.2 Footnote ** LCO 3.2.5.c

DOC L01

R1

4 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ISTS 3.4.1 ACTIONS C and D are specific to RCS cold leg temperature and apply when operation may continue at a reduced RTP for an indefinite period of time. CTS 3.2.5 Actions for RCS cold leg temperature are the same as the Actions for pressurizer pressure and RCS flow rate which require a parameter not within limits be restored to within limits in 2 hours or be in MODE 2 within 6 hours. To reflect the current licensing basis, Condition A is revised to state, “RCS DNB parameter(s) not within limits,” Condition B deletes the phrase “of Condition A,” and ACTIONS C and D are deleted.

4. ISTS LCO 3.4.1 states that the RCS DNB parameters for pressurizer pressure, cold leg temperature, and RCS total flow rate shall be within the limits specified in the COLR. CTS LCO 3.2.5 states that the RCS total flow rate “shall be ≥ 375,000 gpm.” The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. However, NRC-approved Topical Report WCAP-14483, "Generic Methodology for Expanded Core Operating Limits Report," determined that the specific values for the DNB parameters may be relocated to the COLR as long as the limiting RCS total flow limit is retained in the LCO. Therefore, to reflect the CTS 3.2.5 current licensing basis, ISTS LCO 3.4.1 is changed to state that RCS total flow rate “shall be ≥ 375,000 gpm.”

5. The Note to ISTS SR 3.4.1.3 is deleted because it is unnecessary. SR 3.0.1 requires Surveillance to be met during the MODES or other specified conditions in the Applicability. Since ISTS LCO 3.4.1 is only applicable in MODE 1, there is no reason to modify SR 3.4.1.3 with a Note stating that the SR is only required to be met in MODE 1. This change is consistent with ISTS SR 3.4.1.3 in NUREG-1431, which does not contain this Note.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

2

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow [Departure from Nucleate Boiling (DNB)] Limits

2

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on departure from nucleate boiling (DNB) related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO limits for minimum and maximum RCS pressures as measured at the pressurizer are consistent with operation within the nominal operating envelope and are bounded by those used as the initial pressures in the analyses.

The LCO limits for minimum and maximum RCS cold leg temperatures are consistent with operation at the indicated power level and are bounded by those used as the initial temperatures in the analyses.

The LCO limits for minimum RCS flow rate is bounded by the initial flow rate in the analyses. The RCS flow rate is not expected to vary during plant operation with all pumps running.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ [1.3]. This is the acceptance limit for the RCS DNB parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed for include loss of coolant flow events and dropped or stuck control element assembly (CEA) events. A key assumption for the analysis of these events is that the core power distribution is within the limits of [LCO 3.1.6, "Regulating CEA Insertion Limits," LCO 3.1.7, "Part Length CEA Insertion Limits," LCO 3.2.3, "AZIMUTHAL POWER TILT (Tq)," and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)"]. The safety analyses are performed over the following range of initial values: RCS pressure [1785-2400] psig, core inlet temperature [500-580]°F, and reactor vessel inlet coolant flow rate > [95]%.

2

1

2

4

3

use

2225 - 2250 psia

2

535 - 551

≥ 375,000 gpm

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

1

2

BASES

LCO

This LCO specifies limits on the monitored process variables - RCS pressurizer pressure, RCS cold leg temperature, and RCS total flow rate - to ensure that the core operates within the limits assumed for the plant safety analyses. The~~se variables~~ are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

values for RCS pressurizer pressure and RCS cold leg temperature

4

The LCO numerical values for pressure, temperature, and flow rate ~~specified in the COLR~~ are given for the measurement location but have not been adjusted for instrument error. Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of this LCO.

4

APPLICABILITY

In MODE 1, the limits on RCS pressurizer pressure, RCS cold leg temperature, and RCS flow rate must be maintained during steady state operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough so that DNBR is not a concern.

A Note has been added to indicate the limit on pressurizer pressure may be exceeded during short term operational transients such as a THERMAL POWER ramp increase of > 5% RTP per minute or a THERMAL POWER step increase of > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core Safety Limits." Those limits are less restrictive than the limits of this LCO, but violation of SLs merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator should check whether or not an SL may have been exceeded.

ACTIONS

A.1

s

Pressurizer pressure ~~is a~~ controllable and measurable parameter. RCS flow rate is not a controllable parameter and is not expected to vary during steady state operation. With ~~either~~ parameter not within the LCO limits, action must be taken to restore the out of limit parameter.

and RCS cold leg temperature are

a

4

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause of the off normal condition, and to restore the readings within limits. The Completion Time is based on plant operating experience that shows the parameter can be restored in this time period.

2

BASES

ACTIONS (continued)

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

Six hours is a reasonable time that permits the plant power to be reduced at an orderly rate in conjunction with even control of steam generator (SG) heat removal.

~~C.1~~

~~Cold leg temperature is a controllable and measurable parameter. If this parameter is not within the LCO limits, action must be taken to restore the parameter.~~

4

~~The 2 hour Completion Time is based on plant operating experience that shows that the parameter can be restored in this time period.~~

~~D.1~~

~~If Required Action C.1 is not met within the associated Completion Time, THERMAL POWER must be reduced to ≤ [30%] RTP. Plant operation may continue for an indefinite period of time in this condition. At the reduced power level, the potential for violation of the DNB limits is greatly reduced.~~

4

~~The 6 hour Completion Time is a reasonable time that permits power reduction at an orderly rate in conjunction with even control of SG heat removal.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

2

~~[Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure that the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and verify operation is within safety analysis assumptions~~.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
]

SR 3.4.1.2

[Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for cold leg temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
]

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

[The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour Frequency has been shown by operating experience to be sufficient to assess for potential degradation and to verify operation is within safety analysis assumptions.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

This SR is modified by a Note that only requires performance of this SR in MODE 1. The Note is necessary to allow measurement of RCS flow rate at normal operating conditions at power with all RCPs running.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance. This allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow rate is within the bounds of the analyses.

[The Frequency of [18] months reflects the importance of verifying flow after a refueling outage where the core has been altered, which may have caused an alteration of flow resistance.

2

BASES

SURVEILLANCE REQUIREMENTS (continued)

2

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

]

5

The SR is modified by a Note that states the SR is only required to be performed [24] hours after ≥ [90]% RTP. The Note is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance cannot be performed in MODE 2 or below, and will not yield accurate results if performed below 90% RTP.

2

REFERENCES 1. FSAR, Section [15].

2

U

Chapter 15

1

2

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow [Departure from Nucleate Boiling (DNB)] Limits

2

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on departure from nucleate boiling (DNB) related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO limits for minimum and maximum RCS pressures as measured at the pressurizer are consistent with operation within the nominal operating envelope and are bounded by those used as the initial pressures in the analyses.

The LCO limits for minimum and maximum RCS cold leg temperatures are consistent with operation at the indicated power level and are bounded by those used as the initial temperatures in the analyses.

The LCO limits for minimum RCS flow rate is bounded by the initial flow rate in the analyses. The RCS flow rate is not expected to vary during plant operation with all pumps running.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ [1.3]. This is the acceptance limit for the RCS DNB parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed for include loss of coolant flow events and dropped or stuck control element assembly (CEA) events. A key assumption for the analysis of these events is that the core power distribution is within the limits of [LCO 3.1.6, "Regulating CEA Insertion Limits," LCO 3.1.7, "Part Length CEA Insertion Limits," LCO 3.2.3, "AZIMUTHAL POWER TILT (Tq)," and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)"]. The safety analyses are performed over the following range of initial values: RCS pressure [1785-2400] psig, core inlet temperature [500-580]°F, and reactor vessel inlet coolant flow rate > [95]%.

2

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use

2225 – 2250 psia

535 - 551

≥ 375,000 gpm

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

1

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BASES

LCO

This LCO specifies limits on the monitored process variables - RCS pressurizer pressure, RCS cold leg temperature, and RCS total flow rate - to ensure that the core operates within the limits assumed for the plant safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

values for RCS pressurizer pressure and RCS cold leg temperature

4

The LCO numerical values for pressure, temperature, and flow rate specified in the COLR are given for the measurement location but have not been adjusted for instrument error. Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of this LCO.

4

APPLICABILITY

In MODE 1, the limits on RCS pressurizer pressure, RCS cold leg temperature, and RCS flow rate must be maintained during steady state operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough so that DNBR is not a concern.

A Note has been added to indicate the limit on pressurizer pressure may be exceeded during short term operational transients such as a THERMAL POWER ramp increase of > 5% RTP per minute or a THERMAL POWER step increase of > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in Safety Limit (SL) 2.1.1, "Reactor Core Safety Limits." Those limits are less restrictive than the limits of this LCO, but violation of SLs merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator should check whether or not an SL may have been exceeded.

ACTIONS

A.1

s

Pressurizer pressure is a controllable and measurable parameter. RCS flow rate is not a controllable parameter and is not expected to vary during steady state operation. With either parameter not within the LCO limits, action must be taken to restore the out of limit parameter.

and RCS cold leg temperature are

a

4

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause of the off normal condition, and to restore the readings within limits. The Completion Time is based on plant operating experience that shows the parameter can be restored in this time period.

1

2

BASES

ACTIONS (continued)

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

Six hours is a reasonable time that permits the plant power to be reduced at an orderly rate in conjunction with even control of steam generator (SG) heat removal.

~~C.1~~

~~Cold leg temperature is a controllable and measurable parameter. If this parameter is not within the LCO limits, action must be taken to restore the parameter.~~

4

~~The 2 hour Completion Time is based on plant operating experience that shows that the parameter can be restored in this time period.~~

~~D.1~~

~~If Required Action C.1 is not met within the associated Completion Time, THERMAL POWER must be reduced to ≤ [30%] RTP. Plant operation may continue for an indefinite period of time in this condition. At the reduced power level, the potential for violation of the DNB limits is greatly reduced.~~

4

~~The 6 hour Completion Time is a reasonable time that permits power reduction at an orderly rate in conjunction with even control of SG heat removal.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

2

~~[Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure that the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and verify operation is within safety analysis assumptions~~.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

]

SR 3.4.1.2

[Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for cold leg temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

]

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.3

[The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour Frequency has been shown by operating experience to be sufficient to assess for potential degradation and to verify operation is within safety analysis assumptions.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

This SR is modified by a Note that only requires performance of this SR in MODE 1. The Note is necessary to allow measurement of RCS flow rate at normal operating conditions at power with all RCPs running.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance. This allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow rate is within the bounds of the analyses.

[The Frequency of [18] months reflects the importance of verifying flow after a refueling outage where the core has been altered, which may have caused an alteration of flow resistance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

The SR is modified by a Note that states the SR is only required to be performed [24] hours after ≥ [90]% RTP. The Note is necessary to allow measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance cannot be performed in MODE 2 or below, and will not yield accurate results if performed below 90% RTP.

REFERENCES 1. FSAR, Section [15].

U

Chapter 15

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.1, BASES, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS references LCO 3.1.7, “Part Length CEA Insertion Limits.” Part length CEAs are no longer a design feature at PSL Unit 1 and Unit 2. To reflect the current licensing basis, the LCO 3.1.7 reference is deleted. The ISTS includes ISTS 3.2.1, Linear Heat Rate (LHR), ISTS 3.2.2, Total Planar Radial Peaking Factor, ISTS 3.2.3, Total Integrated Radial Peaking Factor, ISTS 3.2.4, Azimuthal Power Tilt, and ISTS 3.2.5, Axial Shape Index (ASI). CTS do not include a Specification for ISTS 3.2.2, Total Planar Radial Peaking Factor. Therefore, the ISTS is renumbered. ISTS 3.2.3, Total Integrated Radial Peaking Factor, ISTS 3.2.4, Azimuthal Power Tilt, and ISTS 3.2.5, Axial Shape Index, are renumbered as ITS 3.2.2, ITS 3.2.3, and ITS 3.2.4, respectively. Therefore, the ISTS Bases is changed to LCO 3.2.4, Axial Shape Index (ASI).

4. Changes have been made to be consistent with changes made to the Specifications.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.1, RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

3.4.2, RCS Minimum Temperature for Criticality

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.2

~~3.1.1.5~~ ~~The~~ Each Reactor Coolant System ~~lowest operating~~ average loop temperature (T_{avg}) shall be ≥ 515°F ~~when the reactor is critical~~.

A01

Applicability

APPLICABILITY: MODE~~S~~ 1 ~~and~~ 2#.

MODE

with T_{avg} in one or more RCS loops < 525°F and

with T_{avg} in one or more RCS loops < 525°F,

ACTION:

A02

ACTION A

With a Reactor Coolant System operating loop temperature (T_{avg}) < 515°F, ~~restore T_{avg} to within its limit within 15 minutes or~~ be in ~~HOT STANDBY~~ within ~~the next~~ ~~15~~ 30 minutes.

A03

MODE 2 with Keff < 1.0

A04

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

~~4.1.1.5~~ The Reactor Coolant System temperature (T_{avg}) shall be determined to be ≥ 515°F.

~~a. Within 15 minutes prior to achieving reactor criticality, and~~

L01

~~b.~~ In accordance with the Surveillance Frequency Control Program when the reactor is critical and the Reactor Coolant System temperature (T_{avg}) is < 525°F.

Applicability

~~With~~ K_{eff} ≥ 1.0.

A01

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.2

~~3.1.1.5~~ ~~The~~ Each Reactor Coolant System ~~lowest operating~~ average loop temperature (T_{avg}) shall be greater than or equal to 515°F.

A01

Applicability

APPLICABILITY: MODE~~S~~ 1 ~~and~~ 2#.

MODE

with T_{avg} in one or more RCS loops < 525°F and

with T_{avg} in one or more RCS loops < 525°F,

ACTION:

A02

ACTION A

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 515°F , ~~restore T_{avg} to within its limit within 15 minutes or~~ be in ~~HOT STANDBY~~ within ~~the next~~ ~~15~~ minutes.

A03

A04

MODE 2 with Keff < 1.0

30

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

~~4.1.1.5~~ The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 515°F:

~~a. Within 15 minutes prior to achieving reactor criticality, and~~

L01

~~b.~~ In accordance with the Surveillance Frequency Control Program when the reactor is critical and the Reactor Coolant System T_{avg} is less than 525°F.

Applicability

~~With~~ K_{eff} greater than or equal to 1.0.

DISCUSSION OF CHANGES
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 4.1.1.5.b requires a determination of RCS average temperature (Tavg) per the Frequency in accordance with the Surveillance Frequency Control Program when the reactor is critical and the RCS Tavg is < 525°F. ITS 3.4.2 Applicability states “MODE 1 with Tavg in one or more RCS loops < 525°F, MODE 2 with Tavg in one or more RCS loops < 525°F and Keff ≥ 1.0.” The ITS couples the monitoring temperature limit with the Applicability. This changes the CTS by moving the temperature monitoring requirement from the Surveillance Frequency to the Applicability. This change is designated as administrative, as it results in no technical change to the CTS.

A03 CTS 3.1.1.5 Action states that with a RCS operating loop temperature (Tavg) < 515°F, "restore (Tavg) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes." ITS 3.4.2, ACTION A, states that with Tavg in one or more RCS loops not within limit, be in MODE 2 with keff < 1.0 within 30 minutes. This changes the CTS by eliminating the redundant and unnecessary requirement to restore Tavg to within its limit within 15 minutes. The change associated with entering MODE 2 with keff < 1.0 instead of HOT STANDBY is discussed in DOC A04. R1

This change is acceptable because it results in no technical change to the Technical Specifications. Although CTS 3.1.1.5 Action appears to only allow 15 minutes to restore the parameter to within the limit, it actually allows the entire 30 minutes to either restore the parameter to within limits or to be in HOT STANDBY (essentially outside the Applicability of CTS 3.1.1.5). In addition, CTS 3.1.1.5 Action only requires actual steps to begin reducing reactor power at the beginning of the last 15 minutes of the 30-minute interval. However, CTS 3.0.2 states that if the LCO is restored prior to expiration of the specified interval, completion of the Action requirements is not required. Therefore, for this case, if the parameter is restored between 15 minutes and 30 minutes after the LCO is not met, completion of the CTS 3.1.1.5 Action to be in HOT STANDBY is not required. Thus, 30 minutes is currently allowed for either the parameter to be restored to within its limit or the unit to be in HOT STANDBY (i.e., only one of the two CTS Actions must be met within 30 minutes). The CTS 3.0.2 requirement is retained in proposed ITS LCO 3.0.2. Therefore, this change does not expand the total time interval allowed to restore the parameter, as a 30-minute interval is already allowed by the CTS. This change is designated as administrative as it results in no technical change to the CTS.

A04 CTS 3.1.1.5 Action states that with a RCS operating loop temperature (Tavg) < 515°F, restore Tavg to within its limit within 15 minutes or "be in HOT STANDBY within the next 15 minutes." ITS 3.4.2, ACTION A, states that with Tavg in one or more RCS loops not within limit, be in MODE 2 with keff < 1.0 within 30 minutes. This changes the CTS from requiring the unit to be in HOT STANDBY to be in MODE 2 with keff < 1.0. Other changes to this CTS Action are discussed in DOC A03.

This change is acceptable because it results in no technical change to the Technical Specifications. CTS 3.1.1.5 is applicable in MODE 1 and MODE 2 with keff ≥ 1.0. CTS 3.0.1 states that compliance with the LCO is required during the MODES or other conditions specified therein, except that upon failure to meet the LCO, the associated Action requirements shall be met. Additionally, CTS 3.0.2 states, in part, that if the LCO is met or no longer applicable prior to expiration of the specified time interval, completion of the actions is not required. Therefore, the CTS 3.1.1.5 Action to be in MODE 3 ceases to be applicable once the unit enters MODE 2 with keff < 1.0, and the Action is exited. As a result, changing the Action to "be in MODE 2 with keff < 1.0," results in no operational difference from the CTS Action. This change is designated as administrative, as it results in no technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.1.1.5.a states that the RCS temperature (Tavg) shall be determined to be ≥ 515°F within 15 minutes prior to achieving reactor criticality. ITS SR 3.4.2.1 requires RCS Tavg in each loop to be verified ≥ 515°F "In accordance with the Surveillance Frequency Control Program when the reactor is critical and the RCS temperature (Tavg) is < 525°F."

The purpose of CTS 4.1.1.5.a is to ensure RCS temperature (Tavg) is within limit when the reactor is critical. The requirement is that RCS Tavg be ≥ 515°F when the unit is operating in MODE 1 and MODE 2 with keff ≥ 1.0. CTS 4.0.4 (ITS SR 3.0.4) requires the SR to be met prior to entry into MODE 2 with keff ≥ 1.0 before

the reactor is critical. Therefore, the RCS Tavg must be determined prior to achieving criticality.

This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of assurance. With RCS temperature (Tavg) < 525°F, CTS 4.0.4 (ITS SR 3.0.4) requires the Surveillance to be performed within 30 minutes (i.e., periodic Frequency in accordance with the Surveillance Frequency Control Program) prior to achieving criticality. This Frequency is adequate to prevent an inadvertent violation of the LCO. In the approach to criticality, the reactor coolant pumps are adding heat to the RCS, so the conditions before and after criticality are similar. The approach to criticality is a carefully controlled evolution where RCS temperature is closely monitored. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.1.1.5

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ ~~[520]~~ 515 °F. (2)

Applicability Footnote # 4.1.1.5.b

APPLICABILITY: MODE 1 with T_{avg} in one or more RCS loops < ~~[535]~~ 525 °F,
MODE 2 with T_{avg} in one or more RCS loops < ~~[535]~~ 525 °F and K_{eff} ≥ 1.0. (2)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.1.5 Actions A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with K_{eff} < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.1.1.5 SR 3.4.2.1	Verify RCS T_{avg} in each loop ≥ ~~[520]~~ 515 °F.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ (2)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.1.1.5

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ ~~[520]~~ 515 °F. 2

Applicability Footnote # 4.1.1.5.b

APPLICABILITY: MODE 1 with T_{avg} in one or more RCS loops < ~~[535]~~ 525 °F,
MODE 2 with T_{avg} in one or more RCS loops < ~~[535]~~ 525 °F and K_{eff} ≥ 1.0. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.1.5 Actions A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with K_{eff} < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.1.1.5 SR 3.4.2.1 Verify RCS T_{avg} in each loop ≥ ~~[520]~~ 515 °F.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for:

a. Operation within the existing instrumentation ranges and accuracies,

b. Operation within the bounds of the existing accident analyses, and

c. Operation with the reactor vessel above its minimum nil ductility reference temperature when the reactor is critical.

The reactor coolant moderator temperature coefficient used in core operating and accident analysis is typically defined for the normal operating temperature range (532°F to ~~573°F~~). The Reactor Protection System receives inputs from the narrow range hot leg temperature and cold detectors, which have a range of ~~520°F to 620°F~~. The RCS loop average temperature (T_{avg}) is controlled using inputs of the same range. Nominal T_{avg} for making the reactor critical is 532°F. Safety and operating analyses for lower temperature have not been made.

580°F

515°F to 665°F and 465°F to 615°F, respectively

1

APPLICABLE SAFETY ANALYSES

There are no accident analyses that dictate the minimum temperature for criticality, but all low power safety analyses assume initial temperatures near the ~~[520]~~°F limit (Ref. 1).

515

2

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of the LCO is to prevent criticality outside the normal operating range (532°F to ~~573~~°F) and to prevent operation in an unanalyzed condition.

580

1 R1

525

The LCO is only applicable below ~~[535]~~°F and provides a reasonable distance to the limit of ~~[520]~~°F. This allows adequate time to trend its approach and take corrective actions prior to exceeding the limit.

515

2

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODES 1 and 2 only and in accordance with this specification. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1, and MODE 2 when $K_{eff} \geq 1.0$. Coupled with the applicability definition for criticality is a temperature limit. Monitoring is required ~~at or~~ below a T_{avg} of ~~[535]~~°F. ~~The no load temperature of 544°F is maintained by the Steam Dump Control System.~~

525

2 3

BASES

ACTIONS

A.1

515

If T_{avg} is below ~~[520]~~°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with K_{eff} < 1.0 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time reflects the ability to perform this action and to maintain the plant within the analyzed range.

2

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

515

RCS loop average temperature is required to be verified at or above ~~[520]~~°F. ~~[The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

4

~~]~~

REFERENCES

1. FSAR, ~~Section [15]~~.

2

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Chapter 15

1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for:

a. Operation within the existing instrumentation ranges and accuracies,

b. Operation within the bounds of the existing accident analyses, and

c. Operation with the reactor vessel above its minimum nil ductility reference temperature when the reactor is critical.

The reactor coolant moderator temperature coefficient used in core operating and accident analysis is typically defined for the normal operating temperature range (532°F to ~~573°F~~). The Reactor Protection System receives inputs from the narrow range hot leg temperature and cold detectors, which have a range of ~~520°F to 620°F~~. The RCS loop average temperature (T_{avg}) is controlled using inputs of the same range. Nominal T_{avg} for making the reactor critical is 532°F. Safety and operating analyses for lower temperature have not been made.

580°F

515°F to 665°F and 465°F to 615°F, respectively

and cold

1

APPLICABLE SAFETY ANALYSES

There are no accident analyses that dictate the minimum temperature for criticality, but all low power safety analyses assume initial temperatures near the ~~[520]~~°F limit (Ref. 1).

515

2

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of the LCO is to prevent criticality outside the normal operating range (532°F to ~~573~~°F) and to prevent operation in an unanalyzed condition.

580

1 R1

525

The LCO is only applicable below ~~[535]~~°F and provides a reasonable distance to the limit of ~~[520]~~°F. This allows adequate time to trend its approach and take corrective actions prior to exceeding the limit.

515

2

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODES 1 and 2 only and in accordance with this specification. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1, and MODE 2 when $K_{eff} \geq 1.0$. Coupled with the applicability definition for criticality is a temperature limit. Monitoring is required ~~at or~~ below a T_{avg} of ~~[535]~~°F. ~~The no load temperature of 544°F is maintained by the Steam Dump Control System.~~

525

2 3

BASES

ACTIONS

A.1

515

If T_{avg} is below ~~[520]~~°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with K_{eff} < 1.0 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time reflects the ability to perform this action and to maintain the plant within the analyzed range.

2

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

515

RCS loop average temperature is required to be verified at or above ~~[520]~~°F. ~~[The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

4

~~]~~

REFERENCES

1. FSAR, ~~Section [15]~~.

2

U

Chapter 15

1

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.2, BASES, RCS MINIMUM TEMPERATURE FOR CRITICALITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS Applicability contains reference to no load temperature maintained by the Steam Bypass Control System or Atmospheric Dump Valves. This information is not related to the ITS 3.4.2 Applicability for the RCS minimum temperature for criticality and is deleted.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

3.4.3, RCS Pressure and Temperature (P/T) Limits

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

LA01

3.4.3-1 and 3.4.3-2

, and RCS heatup and cooldown rates

LCO 3.4.3 ~~3.4.9.1~~ ~~The~~ Reactor Coolant System ~~(except the pressurizer)~~ temperature and pressure shall be ~~limited in accordance with~~ the limit ~~line~~s shown on Figures ~~3.4-2a and 3.4-2b~~ ~~during heatup, cooldown, criticality, and inservice leak and hydrostatic testing~~.

maintained within

A02

Applicability **APPLICABILITY**: At all times.*

A03

Add proposed Conditions A and C Notes

M02

ACTION:

parameter(s)

ACTIONS A and C With any of the above limits exceeded, restore ~~the temperature and/or pressure~~ to within the limits within 30 minutes; ~~perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System;~~ determine that the Reactor Coolant System remains acceptable for continued operations or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~reduce the RCS T_{avg} to less than 200°F~~ within ~~the following~~ ~~30~~ hours ~~in accordance with Figure 3.4-2b~~. R1

LA02

ACTION B

Be in MODE 5

36

MODE 3

within 72 hours

M01

* During hydrostatic testing operations above system design pressure, a maximum temperature change in any one-hour period ~~shall be limited to~~ 5°F.

LCO 3.4.3

A04

of ≤

Shall be maintained

Add proposed ACTION C

M02

with RCS pressure within the limits specified in Figure 3.4.3-2

A01

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

~~4.4.9.1~~

Verify | shown in Figures 3.4.3-1 and 3.4.3-2 | , and RCS heatup and cooldown rates

SR 3.4.3.1

~~a.~~ ~~The~~ Reactor Coolant System temperature and pressure ~~shall be determined to be~~ within the limits in accordance with the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

SR 3.4.3.1 Note

~~b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.~~

L02

~~c. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties as required by 10 CFR 50 Appendix H. The results of these examinations shall be used to update Figures 3.4-2a and 3.4-2b.~~

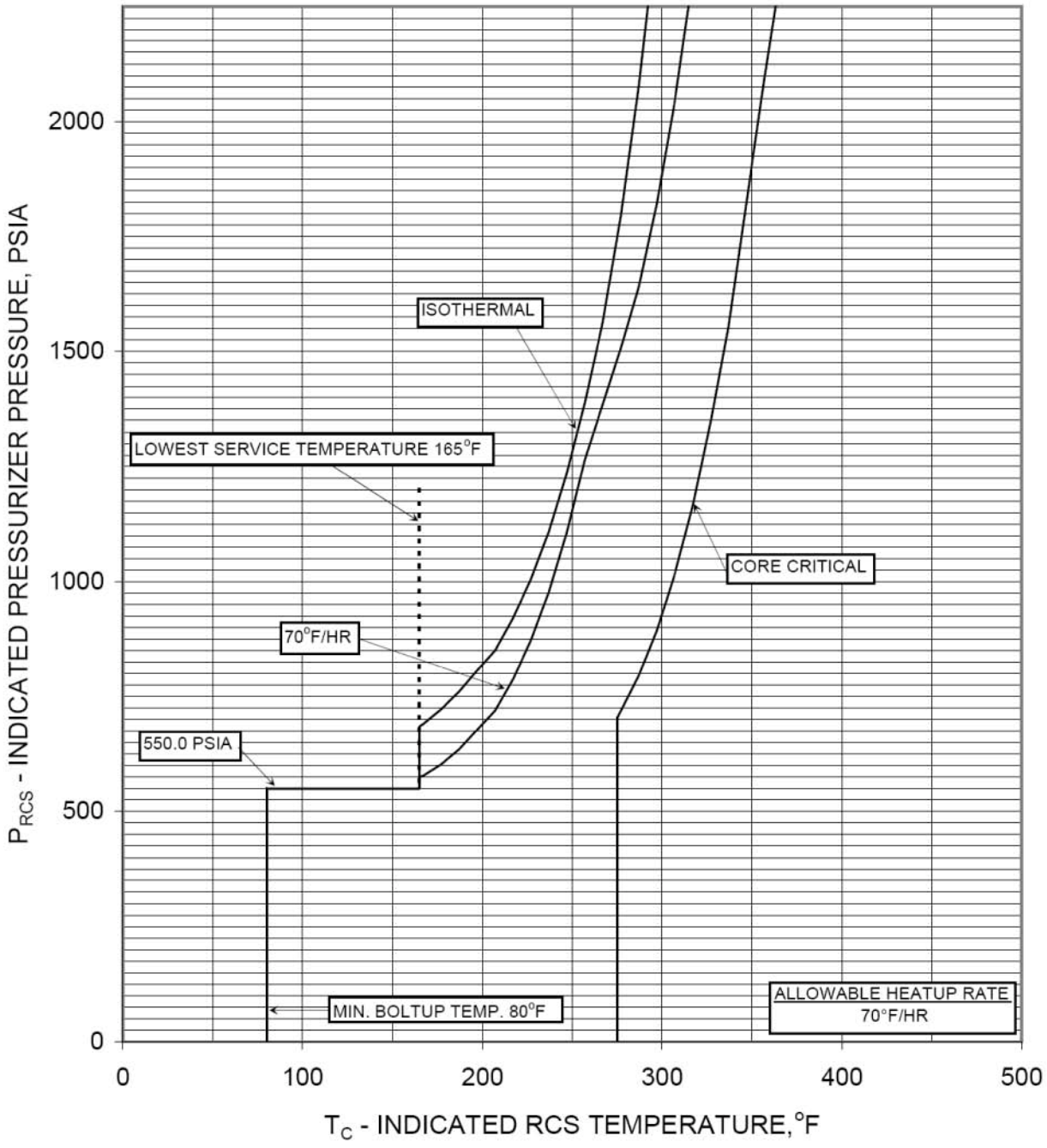
L01

A01

FIGURE ~~3.4-2a~~ 3.4.3-1

~~ST. LUCIE UNIT 1~~ P/T LIMITS, 54 EFPY
HEATUP AND ~~CORE~~ CRITICAL ITY LIMITS

INSERT 1



~~Limiting Material: Lower Shell Axial Welds (Ht. #305424)~~
~~Limiting ART Values at 54 EFPY:~~ ~~1/4T, 210°F~~
~~3/4T, 156°F~~

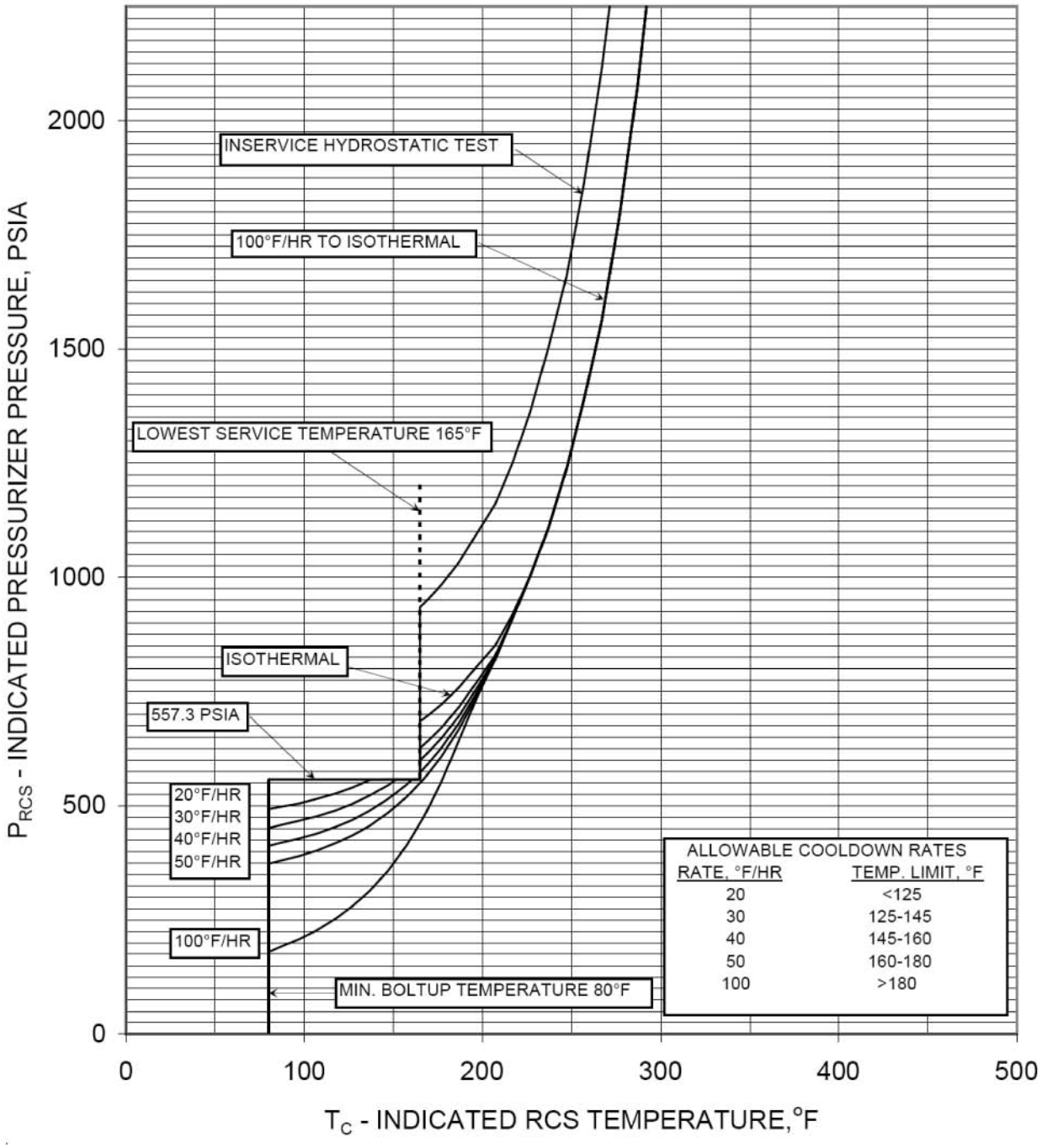
L03

A01

FIGURE ~~3.4-2b~~ 3.4.3-2

~~ST. LUCIE UNIT 1~~ P/T LIMITS, 54 EFPY
COOLDOWN AND INSERVICE TEST LIMITS

INSERT 1



~~Limiting Material: Lower Shell Axial Welds (Ht. #305424)~~
~~Limiting ART Values at 54 EFPY:~~ ~~1/4T, 210°F~~
~~3/4T, 156°F~~

L03

~~DELETED~~

A01

~~DELETED~~

A01

~~DELETED~~

A01

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

Pages 3/4 4-28 through 3/4 4-55 (Amendment No. 90), and Pages 3/4 4-56 through 3/4 4-57 (Amendment No. 80) have been deleted from the Technical Specifications. The next page is 3/4 4-58.

A01

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

LA01

3.4.3-1 and 3.4.3-2

, and RCS heatup and cooldown rates

LCO 3.4.3 ~~3.4.9.1~~ The Reactor Coolant System ~~(except the pressurizer)~~ temperature and pressure ~~shall be limited in accordance with~~ the limit ~~line~~s shown on Figures ~~3.4-2 and 3.4-3~~ ~~during heatup, cooldown, criticality, and inservice leak and hydrostatic testing~~.

maintained within

A02

Applicability **APPLICABILITY**: At all times.

ACTION:

Add proposed Conditions A and C Notes

A03

M02

parameter(s)

ACTIONS A and C With any of the above limits exceeded, restore ~~the temperature and/or pressure~~ to within the limits within 30 minutes; ~~perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System;~~ determine that the Reactor Coolant System remains acceptable for continued operations or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~reduce the RCS T_{avg} to less than 200°F~~ within ~~the next~~ ~~30~~ 36 hours ~~in accordance with Figure 3.4-3~~.

LA02

within 72 hours

M01

ACTION B

Be in MODE 5

R1

with RCS pressure within the limits specified in Figure 3.4.3-2

Add proposed ACTION C

M02

SURVEILLANCE REQUIREMENTS

Verify

shown in Figures 3.4.3-1 and 3.4.3-2

SR 3.4.3.1 ~~4.4.9.1.1~~ ~~The~~Reactor Coolant System temperature and pressure ~~shall be determined to be~~ within the limits in accordance with the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

SR 3.4.3.1 Note

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

~~4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50 Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.~~

L01

A01

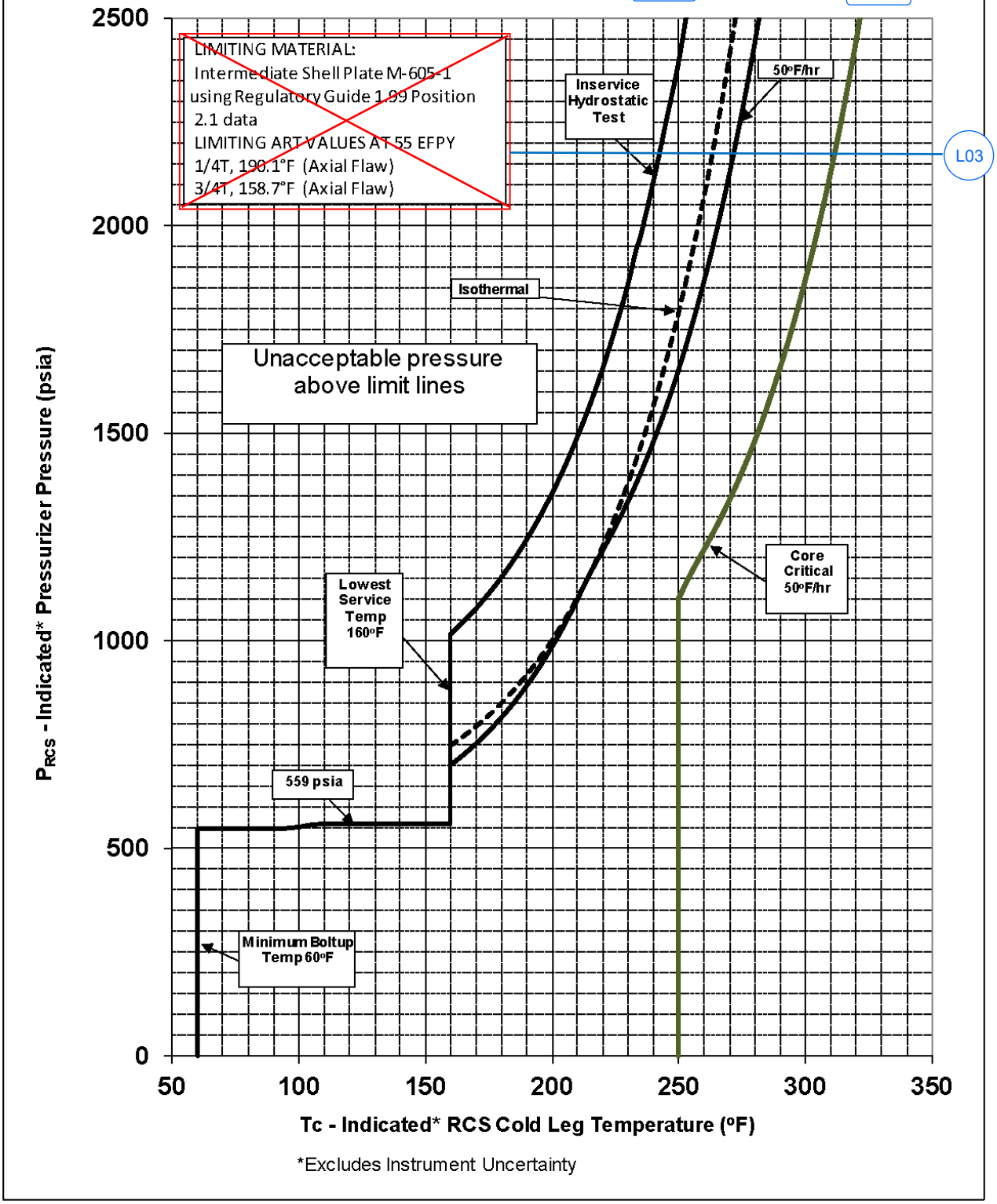
FIGURE ~~3.4-2~~ 3.4.3-1

INSERT 1

~~ST. LUCIE UNIT 2~~ REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 55 EFPY, HEATUP, ~~CORE~~ CRITICAL, AND INSERVICE TEST

ITY

LIMITS



*Excludes Instrument Uncertainty

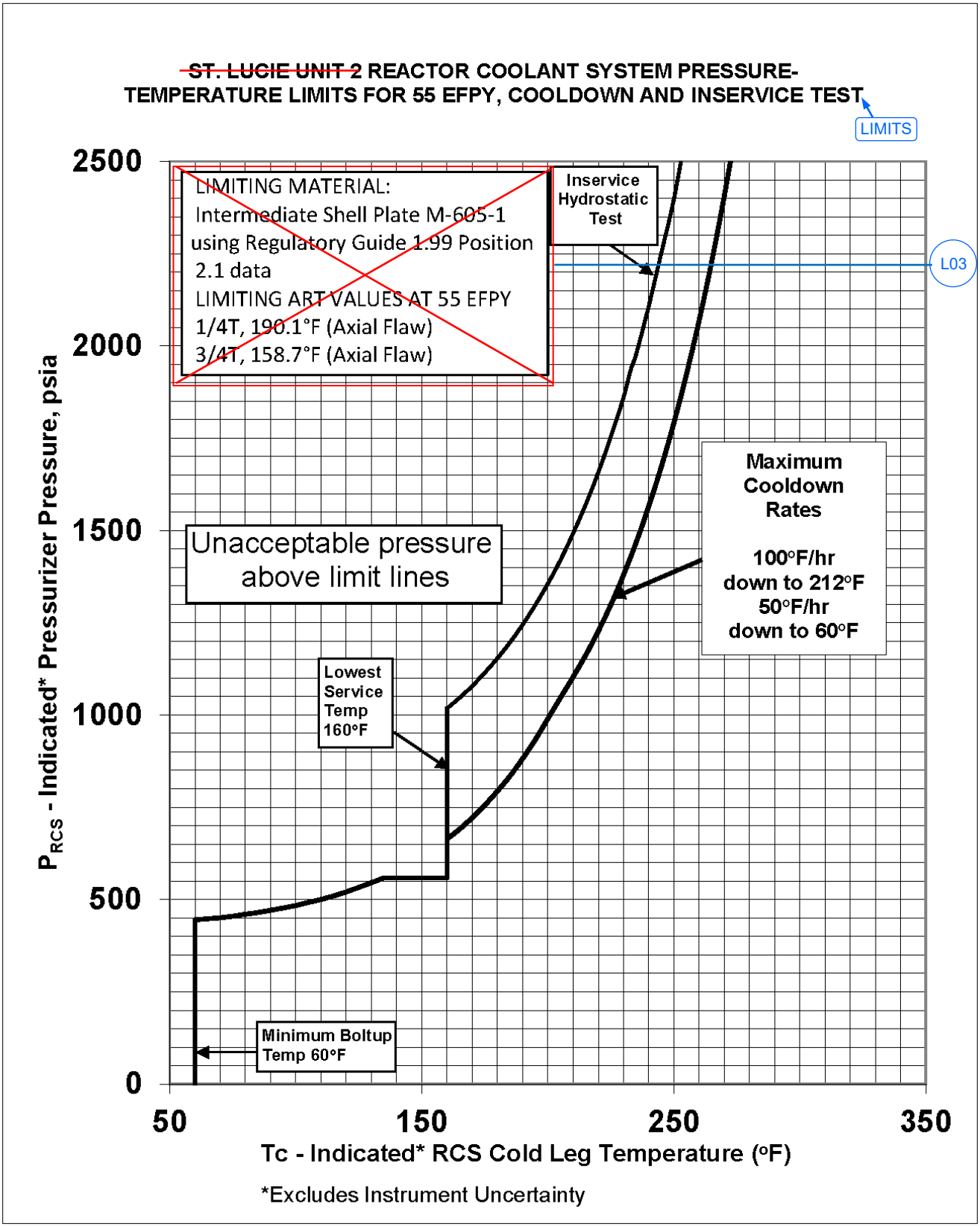
A01

FIGURE ~~3.4-3~~ 3.4.3-2

INSERT 1

~~ST. LUCIE UNIT 2~~ REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 55 EFPY, COOLDOWN AND INSERVICE TEST

LIMITS



A01

~~DELETED~~

A01

~~DELETED~~

DISCUSSION OF CHANGES
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.9.1 requires that the RCS temperature and pressure shall be limited in accordance with the limit lines shown in the applicable CTS figures “during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.” CTS 3.4.9.1 is applicable “at all times.” ITS 3.4.3 states that the RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in Figures 3.4.3-1 and 3.4.3-2. ITS 3.4.3 is applicable "at all times". The CTS LCO is revised to conform to the ISTS. This changes the CTS by eliminating the LCO requirement that the limits must be met during heatup, cooldown, and inservice leak and hydrostatic testing.

This change is acceptable because the limits protecting the reactor vessel and RCS are applicable at all times, including heatup, cooldown, and inservice leak and hydrostatic testing. The RCS pressure and temperature limits are not Mode dependent but are always required to assure the safety of the reactor vessel without exception. Stating that the limits are applicable during heatup, cooldown, and inservice leak and hydrostatic testing in the LCO is not required with the applicability of "at all times" which encompasses the more specific and limited CTS conditions. The applicability provides assurance that the RCS is maintained within the required limits and that the plant will continue to be operated in a safe manner. The list of conditions is duplicative of the conditions stated in the Surveillance. This allowance is more appropriately retained in the Note to SR 3.4.3.1 stating that the SR is only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A03 CTS 3.4.9.1 Action states, in part, that with any of the P/T limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes, perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System, and determine that the Reactor Coolant System remains acceptable for continued operations. ITS 3.4.3, Condition A and Condition C are modified by a Note which requires the determination that the RCS is acceptable for continued operation to be performed whenever the Condition is entered. This changes the CTS by explicitly stating that a determination that the RCS is acceptable for continued operation must be performed whenever the condition is entered. Other changes to the Actions are described in other DOCs. This change is acceptable because it is the current understanding and application of the CTS Action. The CTS 3.4.9.1 Action is currently interpreted as requiring a determination that the RCS is

acceptable for continued operation whenever the LCO is not met. This change is designated as administrative as it clarifies the current understanding of the CTS requirement while providing an "unless otherwise stated," to the requirements of CTS 3.0.2 (ITS LCO 3.0.2).

A04 **Unit 1 only:** CTS 3.4.9.1 Applicability is modified by a footnote that states "During hydrostatic testing operations above system design pressure, a maximum temperature change in any one hour period shall be limited to 5°F." ITS 3.4.3 does not contain this Note. The Note is reworded as "A maximum temperature change of ≤ 5°F in any one hour period during hydrostatic testing above system design pressure" and added statement to ITS LCO 3.4.3. This change is designated as an administrative change and is acceptable because it does do not result in technical changes to the CTS. | R1

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.9.1 Action states, in part, that with any of the P/T limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes, perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System, and determine that the Reactor Coolant System remains acceptable for continued operations. ISTS 3.4.3 Required Action A.2 requires determination that RCS is acceptable for continued operation within 72 hours. The CTS is revised to incorporate the 72 hour Completion Time. This changes the CTS by requiring determination that RCS is acceptable for continued operation within 72 hours.

The change is acceptable because besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the reactor coolant pressure boundary (RCPB) integrity remains acceptable and must be completed before continuing operation. The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. CTS and ISTS provide shutdown Actions if the RCS is not determined to be acceptable for continued operation. This change is designated as more restrictive because it establishes a Completion Time that was not in the CTS.

M02 CTS 3.4.9.1 Action states, in part, that with any of the P/T limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes, perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System, and determine that the Reactor Coolant System remains acceptable for continued operations. The applicability for these Actions is "at all times." ITS 3.4.3 provides separate Conditions (Conditions A and C) to differentiate between ACTIONS in MODES 1, 2, 3, and 4 and ACTIONS in conditions other than MODES 1, 2, 3, and 4. ITS 3.4.3 ACTION C provides requirements for "immediate" restoration of parameters to within limits and determination that RCS is acceptable for continued operation "prior to entering MODE 4." This changes the CTS by requiring an RCS PT limit violation in conditions other than MODE 1, 2, 3, or 4 to be corrected immediately

and provides a time to complete the determination that the RCS is acceptable for continued operation.

The change is necessary because the actions of this LCO, anytime other than in MODE 1, 2, 3, or 4, consider the premise that a violation of the limits occurred during normal plant maneuvering. The CTS Action would appear to provide a half hour in which pressure and temperature requirements could exceed the limits, even if capable of being returned to within limits. Also, if the parameters are incapable of being restored within the limits within 30 minutes, the existing Action would appear to result in the requirement of a Licensee Event Report, since no additional Actions apply (the unit is already in MODE 5 or below). The Actions are more appropriately presented in ITS 3.4.3 ACTION C. Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Completion Time of "immediately" reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in a short period of time in a controlled manner. Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed before continuing operation. The Completion Time of prior to entering MODE 4 forces the evaluation prior to entering a MODE where temperature and pressure can be significantly increased. This change is designated as more restrictive because it establishes Actions with more restrictive Action Completion Times that are applicable anytime other than in MODE 1, 2, 3, or 4, and that are not in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.9.1 states, that the reactor coolant system (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2a and 3.4 2b during heatup, cooldown, criticality, and inservice leak and hydrostatic testing." ITS 3.4.3 states that RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained with limits shown in Figures 3.4.3-1 and 3.4.3-2. This changes the CTS by moving the statement "(except the pressurizer)" which excludes the pressurizer from the LCO limit, to the Bases.

The removal of these details, which are related to design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The RCS P/T limits are applicable to RCPB ferritic materials associated with 10 CFR 50, Appendix G and ASME Code, Section III, Appendix G. The ITS still retains the RCS P/T limits. Neither the CTS nor the ITS P/T limits apply to the pressurizer. It is the ITS convention to state this detail

in the ITS Bases. This detail of the LCO is not required to be in the Technical Specifications in order to provide adequate protection of the public health and safety. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.4.9 Action states, in part, to perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System and determine that the reactor coolant system remains acceptable for continued operation. ITS 3.4.3 also states to determine the RCS is acceptable for continued operation. This changes the CTS by removing these procedural details on how to determine the RCS is acceptable for continued operation, since details on how to make this determination are provided in the ITS 3.4.3 Bases.

The purpose of the CTS action requirements is to verify that the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. ITS 3.4.3 Bases Background and ACTIONS A.1 and A.2 discussions provide details for evaluating that the RCS is acceptable for continued operation. The removal of these procedural details from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the Action to determine the RCS is acceptable for continued operation. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of procedural detail because information relating to the method of performing an action is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 5 - Deletion of Surveillance Requirement)* Unit 1 CTS 4.4.9.1.c and Unit 2 CTS 4.4.9.1.2 specify "The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties as required by 10 CFR 50 Appendix H. The results of these examinations shall be used to update Figures 3.4-2a and 3.4-2b." ISTS 3.4.3 does not contain a corresponding requirement. The CTS is revised to conform to the ISTS. This changes the CTS by deleting Unit 1 CTS 4.4.9.1.c and Unit 2 CTS 4.4.9.1.2 surveillance requirements.

This change is acceptable because the CTS surveillance is not required to assure the pressure and temperature limits in the LCO are properly updated. The unit is required to remove material irradiation surveillance specimens and generate new pressure and temperature curves in accordance with 10 CFR 50, Appendix H. Therefore, the CTS surveillance is redundant to the applicable federal regulations and is not required to assure compliance with the requirements of the federal regulations. The Code of Federal Regulations provide sufficient assurance that the pressure and temperature limits are updated when required and the remaining TS requirements provide adequate assurance that the plant is operated within those pressure and temperature limits. Therefore, the proposed change does not adversely impact the safe operation of the plant. The proposed change is designated less restrictive because a surveillance that is required in the CTS will not be required in the ITS.

L02 **Unit 1 only:** *(Category 5 - Deletion of Surveillance Requirement)* CTS 4.4.9.1.b specifies that "the Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality." ITS 3.4.3 does not contain a corresponding requirement. The CTS is revised to conform to the ISTS. This changes the CTS by deleting CTS 4.4.9.1.b.

The purpose of the CTS requirement is to ensure the RCS P/T criticality limits are not exceeded during the approach to criticality during a plant startup. The proposed change is acceptable because the remaining TS requirements continue to provide adequate assurance that the required pressure and temperature limits continue to be met. The TS contain a separate requirement (ITS 3.4.2) for the minimum temperature for criticality (515°F). The temperature limit for critical operation provides assurance that critical operation does not occur at temperatures lower than the RCS P/T limits shown in the applicable CTS figure. As stated in the ITS 3.4.3 bases, the RCS minimum temperature for criticality specified in ITS 3.4.2 is more restrictive than the RCS pressure and temperature limits for criticality. Therefore, the performance of CTS surveillance 4.4.9.1.b is not required to assure the plant is operated within the required pressure and temperature limits. The RCS minimum temperature for criticality requirements provide sufficient assurance that the plant is operated within the RCS pressure and temperature limits. The proposed change is designated as less restrictive because a surveillance that is required by the CTS will not be required in the ITS.

L03 **Unit 1 only**: (*Category 1 – Relaxation of LCO Requirements*) CTS Figures 3.4-2a and 3.4-2b describe the limiting material and limiting ART values at 54 EFPY used to determine the P/T limits. The ITS Figures 3.4.3-1 and 3.4.3-2 do not include this information. This changes the CTS by deleting this information from the CTS.

The purpose of this information is to provide additional detail as to how the P/T curves were generated. However, deleting this information is acceptable because it is not necessary to be listed in the Figures to use the Figures effectively. ITS 3.4.3 requires the Figure limits to be met. The details as to how the Figures are generated is not needed to comply with the LCO. The ITS 3.4.3 Bases describes that the Figures were generated to comply with the applicable regulatory requirements of 10 CFR 50, Appendix G and ASME Section III,

Appendix G. Therefore, since PSL is required to comply with 10 CFR 50, Appendix G and ASME Section III, Appendix G, this additional information is not required. This change is designated as less restrictive because certain details related to how the LCO was generated are being deleted.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9.1

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in ~~the PTLR~~.

Figures 3.4.3-1 and 3.4.3-2

A maximum temperature change of ≤ 5°F in any one hour period shall be maintained during hydrostatic testing operations above system design pressure.

3

4

Applicability

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.9.1 Action A. ----------NOTE---------- Required Action A.2 shall be completed whenever this Condition is entered. ---------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
3.4.9.1 Action B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5 with RCS pressure ~~< [500] psig~~.	6 hours 36 hours
DOC M02 C. ----------NOTE---------- Required Action C.2 shall be completed whenever this Condition is entered. ---------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

within the limits specified in Figure 3.4.3-2

5

R1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.4.9.1 SR 3.4.3.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates within limits specified in ~~the PTLR~~. Figures 3.4.3-1 and 3.4.3-2	~~[30 minutes~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2
3

INSERT 1

3

Insert 1 (2)

CTS Figure 3.4-2a

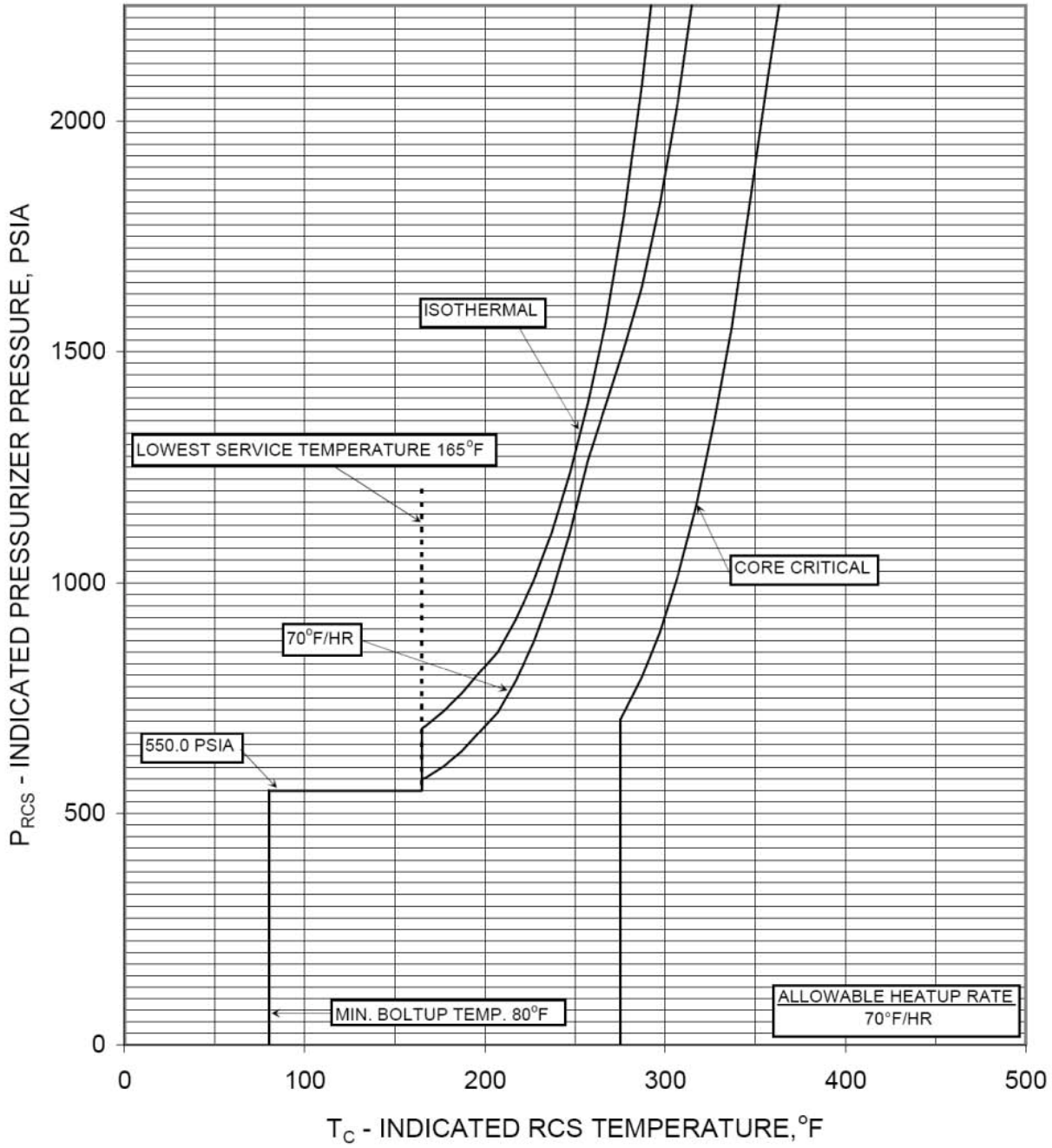


Figure 3.4.3-1
Reactor Coolant System Pressure versus Temperature Limits -
Heatup and Criticality Limits
(54 EFPY)

Insert 1 (2)

CTS Figure 3.4-2b

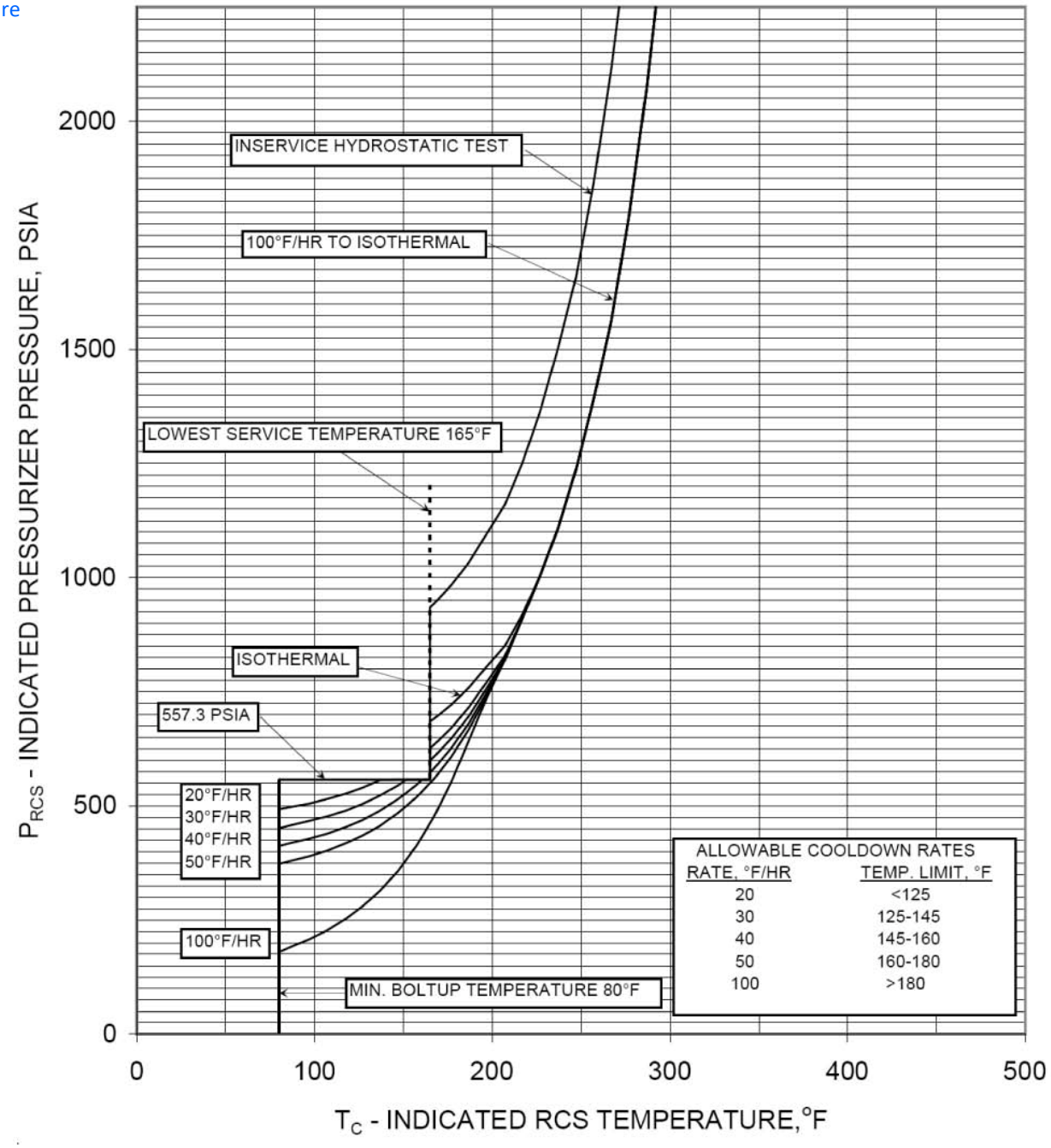


Figure 3.4.3-2
Reactor Coolant System Pressure versus Temperature Limits -
Cooldown and Inservice Test Limits
(54 EFPY)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9.1

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in ~~the PTLR~~.

Figures 3.4.3-1 and 3.4.3-2

3

Applicability

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.9.1 Action A. ----------NOTE---------- Required Action A.2 shall be completed whenever this Condition is entered. ---------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
3.4.9.1 Action B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5 with RCS pressure ~~< [500] psig~~ within the limits specified in Figure 3.4.3-2.	6 hours 36 hours 5 R1
DOC M02 C. ----------NOTE---------- Required Action C.2 shall be completed whenever this Condition is entered. ---------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.4.9.1.1 SR 3.4.3.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates within limits specified in ~~the PTLR~~. Figures 3.4.3-1 and 3.4.3-2	~~[~~ 30 minutes ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2
3

INSERT 1

3

Insert 1 (2)

CTS Figure 3.4-2

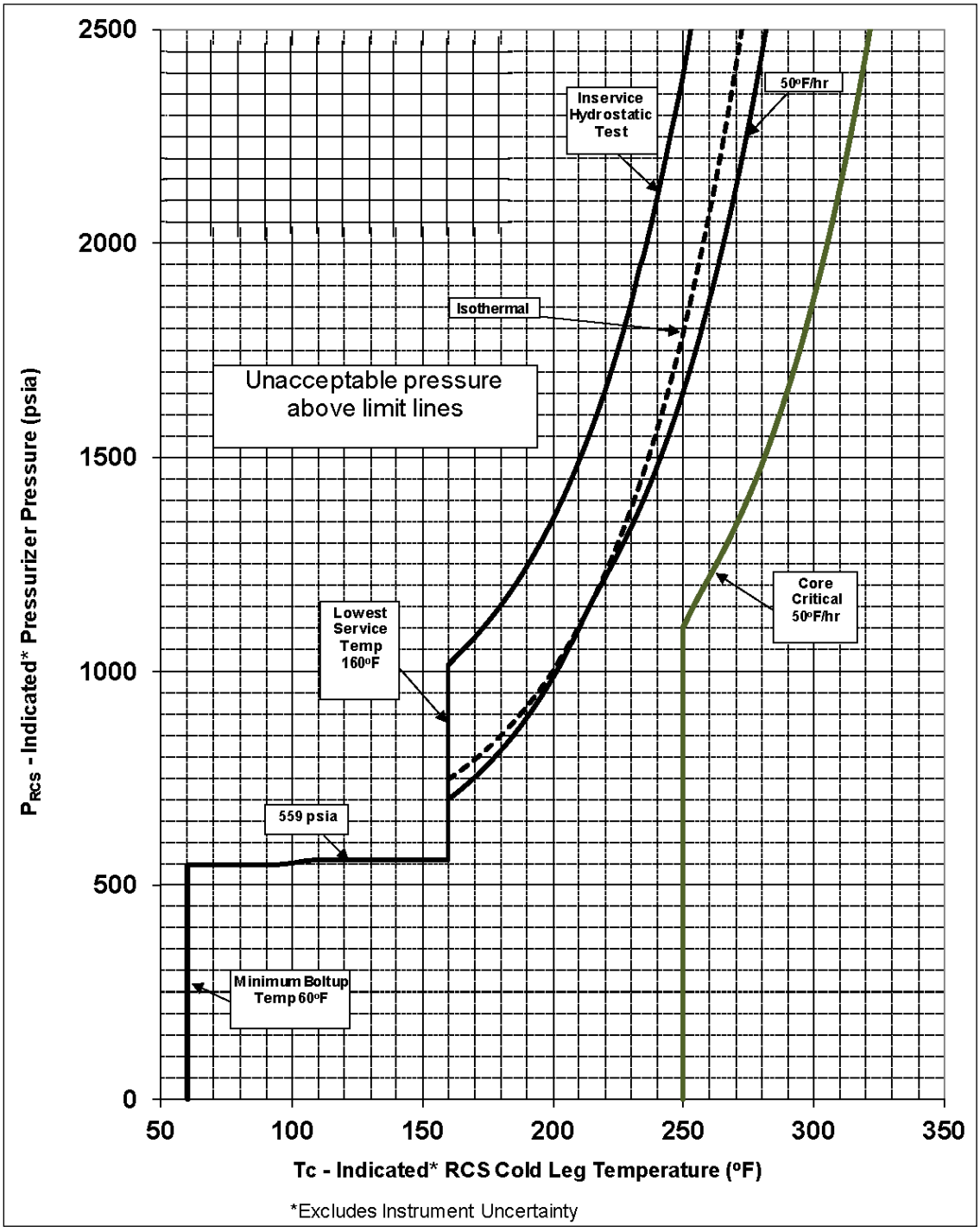


Figure 3.4.3-1
Reactor Coolant System Pressure versus Temperature Limits -
Heatup, Criticality, and Inservice Test Limits
(55 EFPY)

Insert 1 (2)

CTS Figure 3.4-3

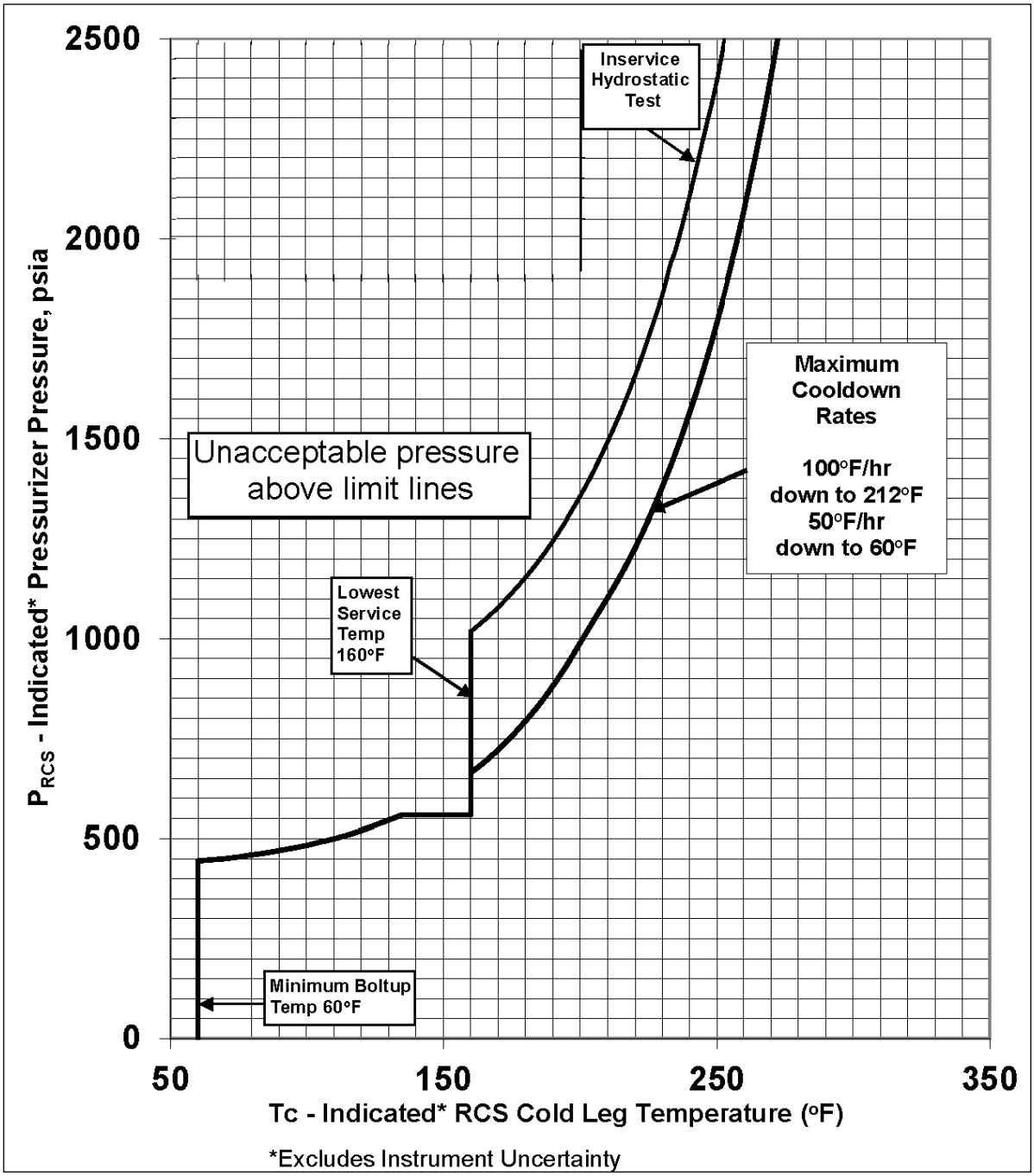


Figure 3.4.3-2
Reactor Coolant System Pressure versus Temperature Limits -
Cooldown and Inservice Test Limits
(55 EFPY)

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. PSL is not adopting a Pressure Temperature Limits Report (PTLR) and is retaining the RCS Pressure and Temperature (P/T) limits in the ITS. Therefore, references to the PTLR have been changed to Figures 3.4.3-1 and 3.4.3-2. In addition, the CTS P/T figures have been added to ITS 3.4.3 as Figures 3.4.3-1 and 3.4.3-2.

4. **Unit 1 only:** ISTS LCO 3.4.3 is modified to include a statement that limits the temperature change allowed during hydrostatic testing operations above system design pressure. The ITS maintains a P/T limit restriction consistent with the equivalent CTS requirement and licensing basis.

5. ISTS 3.4.3, Required Action B.2 is modified to replace the requirement to reduce RCS pressure < [500] psig with the requirement for RCS pressure to be "within the limits specified in Figure 3.4.3-2," consistent with the equivalent CTS requirement and licensing basis. | R1

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.3-1 and 3.4.3-2

the

~~The PTLR~~ contain~~s~~ P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

core critical,

1

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 3).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

BASES

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit includes the Reference 2 requirement that the limit be no less than 40°F above the heatup curve or the cooldown curve and not less than the minimum permissible temperature for the ISLH testing. However, the criticality limit is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) Analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

The RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer.

These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature,

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced), and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.2, "RCS Minimum

BASES

APPLICABILITY (continued)

Temperature for Criticality," and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures.

ACTIONS

A.1 and A.2

Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed ~~before continuing operation~~. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

within 72 hours

1

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

BASES

ACTIONS (continued)

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because:

a. The RCS remained in an unacceptable P/T region for an extended period of increased stress or

b. A sufficiently severe event caused entry into an unacceptable region.

Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in MODE 3 within 6 hours and in MODE 5 with RCS pressure ~~< [500] psig~~ within the limits specified in Figure 3.4.3-2 within 36 hours.

2 | R1

The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

The actions of this LCO, anytime other than in MODE 1, 2, 3, or 4, consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures. Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The Completion Time of "immediately" reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in a short period of time in a controlled manner.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

BASES

ACTIONS (continued)

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The Completion Time of prior to entering MODE 4 forces the evaluation prior to entering a MODE where temperature and pressure can be significantly increased. The evaluation for a mild violation is possible within several days, but more severe violations may require special, event specific stress analyses or inspections.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

RCS P/T

Verification that operation is within the ~~PTLR~~ limits is required when RCS pressure and temperature conditions are undergoing planned changes. ~~[This Frequency of 30 minutes is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 3

~~]~~

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note that requires this SR be performed only during RCS system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. ~~[NRC approved topical report that defines the methodology for determining the P/T limits]~~. WCAP-17197-NP, St. Lucie Unit 1 RCS Pressure and Temperature Limits and Low-Temperature Overpressure Protection Report for 54 Effective Full Power Years, Revision 1, January 2012. (2)
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. ASTM E 185-82, July 1982.
5. 10 CFR 50, Appendix H.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.3-1 and 3.4.3-2

the

~~The PTLR~~ contain~~s~~ P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

core critical,

1

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 3).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

BASES

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit includes the Reference 2 requirement that the limit be no less than 40°F above the heatup curve or the cooldown curve and not less than the minimum permissible temperature for the ISLH testing. However, the criticality limit is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) Analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

The RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer.

These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature,

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced), and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.2, "RCS Minimum

BASES

APPLICABILITY (continued)

Temperature for Criticality," and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures.

ACTIONS

A.1 and A.2

Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed ~~before continuing operation~~. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

within 72 hours

1

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

BASES

ACTIONS (continued)

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because:

a. The RCS remained in an unacceptable P/T region for an extended period of increased stress or

b. A sufficiently severe event caused entry into an unacceptable region.

Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in MODE 3 within 6 hours and in MODE 5 with RCS pressure ~~< [500] psig~~ within the limits specified in Figure 3.4.3-2 within 36 hours. (2) R1

The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

The actions of this LCO, anytime other than in MODE 1, 2, 3, or 4, consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures. Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The Completion Time of "immediately" reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in a short period of time in a controlled manner.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

BASES

ACTIONS (continued)

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The Completion Time of prior to entering MODE 4 forces the evaluation prior to entering a MODE where temperature and pressure can be significantly increased. The evaluation for a mild violation is possible within several days, but more severe violations may require special, event specific stress analyses or inspections.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

RCS P/T

Verification that operation is within the ~~PTLR~~ limits is required when RCS pressure and temperature conditions are undergoing planned changes. ~~[This Frequency of 30 minutes is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 3

~~]~~

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note that requires this SR be performed only during RCS system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. ~~[NRC approved topical report that defines the methodology for determining the P/T limits]~~.

WCAP-18275-NP, St. Lucie Unit 2 Heatup and Cooldown Limit Curves for Normal Operation Through End of License Extension, Revision 0, November 2019. (2)

2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. ASTM E 185-82, July 1982.
5. 10 CFR 50, Appendix H.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.3, BASES, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.3, RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 4

3.4.4, RCS Loops – MODES 1 and 2

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

LCO 3.4.4

3.4.1.1 ~~Both reactor coolant~~ loops ~~and both reactor coolant pumps in each loop~~ shall be in operation.

Two RCS

OPERABLE and

A02

LA01

Applicability

APPLICABILITY: MODES 1 and 2.

ACTION:

ACTION A

~~With less than the above required reactor coolant pumps in operation,~~ be in ~~at least HOT STANDBY~~ within ~~1~~ hour.

Requirements of LCO not met

MODE 3

6

L01

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

~~4.4.1.1~~ ~~The above required reactor coolant~~ loop~~s~~ shall be verified to be in operation ~~and circulating reactor coolant~~ in accordance with the Surveillance Frequency Control Program.

Each RCS

LA01

A01

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

A02

LCO 3.4.4

3.4.1.1 ~~Both Reactor Coolant~~ loops ~~and both Reactor Coolant pumps in each loop~~ shall be in operation.

Two RCS

OPERABLE and

LA01

Applicability

APPLICABILITY: 1 and 2.~~*~~

A03

ACTION:

ACTION A

~~With less than the above required Reactor Coolant pumps in operation~~, be in ~~at least HOT STANDBY~~ within ~~1~~ hour.

Requirements of LCO not met

MODE 3

6

L01

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

4.4.1.1 ~~The above required Reactor Coolant~~ loop~~s~~ shall be verified to be in operation ~~and circulating Reactor Coolant~~ in accordance with the Surveillance Frequency Control Program.

Each RCS

LA01

~~* See Special Test Exception 3.10.3~~

A03

DISCUSSION OF CHANGES
ITS 3.4.4, RCS LOOPS – MODES 1 and 2

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.1.1 states that both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation. ITS 3.4.4. states that two RCS loops shall be OPERABLE and in operation. This changes the CTS by requiring the RCS loops to be OPERABLE. This change is acceptable because it is consistent with the current use and understanding of the LCO. It is not sufficient for an RCS loop to be in operation if it is not capable of performing its safety function (i.e., OPERABLE). This change is designated as administrative as it clarifies the current understanding of a requirement.

A03 **Unit 2 only:** The Applicability of CTS 3.4.1.1 is modified by footnote * that states "See Special Test Exception 3.10.3." The ITS 3.4.4 Applicability does not contain the footnote or a reference to the Special Test Exceptions. This changes the CTS to not include the footnote "See Special Test Exception 3.10.3."

The purpose of the footnote reference is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS. See Unit 2 CTS 3/4.10.3 Discussion of Changes for changes to this Special Test Exception.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.4.1.1 states that both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation. ITS 3.4.4. states that two RCS loops shall be OPERABLE and in operation. CTS 4.4.1.1 states, in part, that the above required reactor coolant loops shall be verified to be in

operation and circulating reactor coolant. ITS SR 3.4.4.1 states verify each RCS loop is in operation.

This changes the CTS by moving the LCO detail of two reactor coolant pumps in each loop, and by moving the Surveillance Requirement to verify that the reactor coolant loops are circulating reactor coolant, to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that two RCS loops be in operation. This will ensure adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power. Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG that is OPERABLE. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.4.1.1 Action states that with less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour. ITS 3.4.4 ACTION A states when the RCS loop requirements are not met, the unit must be in MODE 3 (i.e., HOT STANDBY) within 6 hours. This changes the CTS by relaxing the Completion Time from 1 hour to 6 hours.

The purpose of CTS 3.4.1.1 ACTION is to require a unit shutdown if the necessary reactor coolant loop flow is not available. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the low probability of a DBA occurring during the allowed Completion Time. Operating experience has shown that 6 hours is a reasonable time to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. It is likely that failure to meet the LCO requirements will lead to a reactor trip on low flow. However, if the LCO is not met for a reason that does not lead to a reactor trip, then 6 hours to transition from full power operation to MODE 3 is consistent with the Completion Time provided for a loss of safety function for other systems and with LCO 3.0.3. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODES 1 and 2

3.4.1.1 LCO 3.4.4 Two RCS loops shall be OPERABLE and in operation.

Applicability APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.4.1.1 SR 3.4.4.1 Verify each RCS loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODES 1 and 2

3.4.1.1 LCO 3.4.4 Two RCS loops shall be OPERABLE and in operation.

Applicability APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.4.1.1 SR 3.4.4.1 Verify each RCS loop is in operation.	~~[~~ ~~12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.4, RCS LOOPS – MODES 1 and 2

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission,

b. Improving the neutron economy by acting as a reflector,

c. Carrying the soluble neutron poison, boric acid,

d. Providing a second barrier against fission product release to the environment, and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains a SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with both RCPs in operation in each loop. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying two RCS loops provides the minimum necessary paths (two SGs) for heat removal.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the Design Bases Accident (DBA) initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming four RCPs are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to RCP operation are the four pump coastdown, single pump locked rotor, ~~single pump (broken shaft or coastdown),~~ and rod withdrawal events (Ref. 1). 1

Steady state DNB analysis had been performed for the ~~[~~four~~]~~ pump combination. For ~~[~~four~~]~~ pump operation, the steady state DNB analysis, which generates the pressure and temperature and Safety Limit (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of ~~107~~% RTP. This is the design overpower condition for four pump operation. The ~~107~~% value is the accident analysis setpoint of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit. 2 112 112 1

RCS Loops - MODES 1 and 2 satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG that is OPERABLE. SG, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the Reactor Protection System (RPS) in MODES 1 and 2. A reactor trip places the plant in MODE 3 if any SG level is ≤ ~~[25]~~% as sensed by the RPS. The minimum water level to declare the SG OPERABLE is ~~[25]~~%. narrow range 2 35 35

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

BASES

APPLICABILITY (continued)

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, 5, and 6.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.6, "RCS Loops - MODE 4,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~.

3
3

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits. It should be noted that the reactor will trip and place the plant in MODE 3 as soon as the RPS senses less than four RCPs operating.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which help to ensure that forced flow is providing heat removal while maintaining the margin to DNB. ~~[The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.~~

2

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

REFERENCES 1. FSAR, Section ~~[ ]~~ 15.3 2

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission,

b. Improving the neutron economy by acting as a reflector,

c. Carrying the soluble neutron poison, boric acid,

d. Providing a second barrier against fission product release to the environment, and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains a SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with both RCPs in operation in each loop. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying two RCS loops provides the minimum necessary paths (two SGs) for heat removal.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the Design Bases Accident (DBA) initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming four RCPs are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to RCP operation are the four pump coastdown, single pump locked rotor, ~~single pump (broken shaft or coastdown),~~ and rod withdrawal events (Ref. 1). 1

Steady state DNB analysis had been performed for the ~~[~~four~~]~~ pump combination. For ~~[~~four~~]~~ pump operation, the steady state DNB analysis, which generates the pressure and temperature and Safety Limit (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of 107% RTP. This is the design overpower condition for four pump operation. The 107% value is the accident analysis setpoint of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit. 2

RCS Loops - MODES 1 and 2 satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG that is OPERABLE. SG, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the Reactor Protection System (RPS) in MODES 1 and 2. A reactor trip places the plant in MODE 3 if any SG level is ≤ ~~[25]~~% as sensed by the RPS. The minimum water level to declare the SG OPERABLE is ~~[25]~~%. 35 35 narrow range 2

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

BASES

APPLICABILITY (continued)

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, 5, and 6.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.6, "RCS Loops - MODE 4,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~.

3
3

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits. It should be noted that the reactor will trip and place the plant in MODE 3 as soon as the RPS senses less than four RCPs operating.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which help to ensure that forced flow is providing heat removal while maintaining the margin to DNB. ~~[The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.~~

2

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

REFERENCES 1. FSAR, Section ~~[ ]~~. 15.3 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.4, BASES, RCS LOOPS – MODES 1 and 2

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the Specification Title.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.4, RCS LOOPS – MODES 1 and 2

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

3.4.5, RCS Loops – MODE 3

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

LCO 3.4.5

~~3.4.1.2~~ ~~The~~ Two RCS ~~reactor coolant~~ loops ~~listed below~~ shall be OPERABLE and ~~at least~~ one ~~of these reactor coolant~~ loop~~s~~ shall be in operation.*

RCS

~~a. Reactor Coolant Loop A and its associated steam generator and at least one associated reactor coolant pump.~~

~~b. Reactor Coolant Loop B and its associated steam generator and at least one associated reactor coolant pump.~~

LA01

Applicability

APPLICABILITY: MODE 3.

ACTION:

ACTION A

a. ~~With less than the above required reactor coolant~~ loop~~s~~ OPERABLE, restore ~~the required~~ loop~~s~~ to OPERABLE status within 72 hours or be in ~~HOT~~ ~~SHUTDOWN~~ within the next 12 hours.

one RCS

IN

ACTION B

MODE 4

not

Two RCS loops inoperable OR Required RCS

M01

ACTION C.1

b. ~~With no reactor coolant~~ loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1~~.1~~ ~~and within one (1) hour~~ initiate ~~corrective~~ action to ~~return the required reactor coolant~~ loop to operation.

LCO

ACTION C.2

Immediately

Restore one RCS

OPERABLE status and

M01

M02

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.4.5.3 Note

L01

SR 3.4.5.3

~~4.4.1.2.1~~ ~~At least the above~~ required ~~reactor coolant~~ pump~~s~~, ~~if not in operation, shall be determined to be OPERABLE~~ in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability.

each

RCS

SR 3.4.5.1

~~4.4.1.2.2~~ ~~At least~~ one ~~reactor coolant~~ loop shall be verified to be in operation ~~and circulating reactor coolant~~ in accordance with the Surveillance Frequency Control Program.

LA02

SR 3.4.5.2

~~4.4.1.2.3~~ ~~The required~~ steam generators shall be determined OPERABLE by verifying the secondary side water level to be ≥ 10% ~~of narrow range indication~~ in accordance with the Surveillance Frequency Control Program.

each

removed from operation

≤

per 8 hour period

M03

LCO 3.4.5 Note

* All reactor coolant pumps may be ~~de-energized~~ for ~~up to~~ 1 hour provided (1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1~~.1~~ and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

LCO

A01

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

LCO 3.4.5

~~3.4.1.2~~ ~~The~~ Two RCS ~~Reactor Coolant~~ loops ~~listed below~~ shall be OPERABLE and ~~at least~~ one ~~of these Reactor Coolant~~ loops shall be in operation.*

RCS

~~a. Reactor Coolant Loop 2A and its associated steam generator and at least one associated Reactor Coolant pump.~~

~~b. Reactor Coolant Loop 2B and its associated steam generator and at least one associated Reactor Coolant pump.~~

LA01

Applicability

APPLICABILITY: MODE 3

ACTION:

ACTION A

a. ~~With less than the~~ one RCS ~~above required Reactor Coolant~~ loop~~s~~ IN OPERABLE, restore ~~the required~~ loop~~s~~ to OPERABLE status within 72 hours or be in ~~HOT SHUTDOWN~~ within the next 12 hours.

ACTION B

MODE 4

not

Two RCS loops inoperable OR Required RCS

M01

ACTION C.1

b. ~~With no Reactor Coolant~~ loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1~~.1~~

LCO

ACTION C.2

and immediately initiate ~~corrective~~ action to ~~return the required Reactor Coolant~~ loop to operation.

Restore one RCS

OPERABLE status and

M01

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.4.5.3 Note

L01

SR 3.4.5.3

~~4.4.1.2.1~~ ~~At least the above~~ required ~~Reactor Coolant~~ pump~~s~~, ~~if not in operation, shall be determined to be OPERABLE~~ in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability.

each

RCS

SR 3.4.5.1

~~4.4.1.2.2~~ ~~At least~~ one ~~Reactor Coolant~~ loop shall be verified to be in operation ~~and circulating reactor coolant~~ in accordance with the Surveillance Frequency Control Program.

LA02

SR 3.4.5.2

~~4.4.1.2.3~~ ~~The required~~ steam generator(s) shall be determined OPERABLE verifying the secondary side water level to be ≥ 10% ~~indicated narrow range level~~ in accordance with the Surveillance Frequency Control Program.

each

removed from operation

≤

per 8 hour period

M03

LCO 3.4.5 Note

* All Reactor Coolant pumps may be ~~deenergized~~ for ~~up to~~ 1 hour provided (1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1~~.1~~ and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

LCO

DISCUSSION OF CHANGES
ITS 3.4.5, RCS LOOPS – MODE 3

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.1.2 Action b. states, in part, that with no reactor coolant loop in operation, take the actions specified therein. ITS 3.4.5 ACTION C states the Condition as two RCS loops inoperable or required RCS loop not in operation. This changes the CTS by adding the Condition “two RCS loops inoperable” and a Required Action to immediately initiate action to restore the RCS loops to OPERABLE status.

This change is acceptable because it provides appropriate actions if two RCS loops are inoperable. Under these conditions, immediate action is necessary to ensure certain unit transients do not occur, and action is immediately taken to restore one loop to OPERABLE status. This change is designated as more restrictive, because it requires an additional condition under which the actions must be taken, and an additional action that the RCS loop be restored to OPERABLE status.

M02 **Unit 1 only:** CTS 3.4.1.2 Action b. states, in part, within one (1) hour initiate corrective action to return the required reactor coolant loop to operation. ITS Required Action C.2 states immediately initiate action to restore one RCS loop to OPERABLE status and operation. This changes the CTS by revising the Actions to “immediately” require actions to be taken when a required RCS loop is not in operation. | R1

This change is acceptable because it provides appropriate immediate actions for a required RCS loop not in operation. This action is required to assure continued safe operation. Action to restore one RCS loop to OPERABLE status and operation is necessary to be able to remove the decay heat generated by the reactor. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

M03 CTS 3.4.1.2 states, in part, that the reactor coolant loops listed shall be OPERABLE and at least one of these reactor coolant loops shall be in operation. CTS 3.4.1.2 is modified by a footnote (*) that states, in part, that all reactor coolant pumps may be de energized for up to 1 hour under the conditions specified therein. ITS 3.4.5 contains the same allowance, but limits the use of the 1 hour exception to once per 8 hour period. This changes the CTS by

modifying the 1 hour allowance that all reactor coolant pumps may be de-energized and limits the usage of the allowance to once per 8 hour period.

The purpose of the 1 hour allowance is to allow a reactor coolant loop to be removed from operation to support testing. This change is acceptable because it helps ensure that boron stratification and inadequate decay heat removal do not occur should multiple 1 hour periods be required. This change is designated as more restrictive because it limits the allowance to 1 hour per 8 hour period, and that restriction does not currently exist

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.1.2 states, that the reactor coolant loops listed below shall be OPERABLE and at least one of these reactor coolant loops shall be in operation. CTS 3.4.1.2 contains a description of what constitutes an OPERABLE RCS loop. ITS 3.4.5 does not include a description of what constitutes an OPERABLE RCS loop. This changes the CTS by moving the details of what constitutes an OPERABLE RCS loop, to the Bases.

The removal of these details related to system design from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains a requirement for two RCS loops to be OPERABLE and one RCS loop to be in operation. The removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.2.2 states, in part, that at least one reactor coolant loop shall be verified to be in operation and circulating reactor coolant. ITS SR 3.4.5.1 states verify one RCS loop is in operation. This changes the CTS by moving the detail to verify that the reactor coolant loops are circulating reactor coolant, to the Bases.

The removal of this detail for performing the Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that a RCS loop be in operation. This will require adequate forced flow for core heat removal. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5.

This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.4.1.2.1 states that the required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE by verifying correct breaker alignments and indicated power availability. ITS SR 3.4.5.3 requires verification of correct breaker alignment and indicated power availability to each required pump. ITS SR 3.4.5.3 is modified by a Note that states "Not required to be performed until 24 hours after a required pump is not in operation." This changes the CTS by not requiring the Surveillance Requirement to be performed until 24 hours after a required pump is not in operation.

The purpose of the Surveillance is to ensure that the standby reactor coolant pump is ready to operate. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The Note provides time to perform the Surveillance to verify correct breaker alignment and indicated power availability. Without the Note, the Surveillance may not be performed within its Frequency immediately after taking a pump out of operation requiring the Surveillance to be declared not met per SR 3.0.1 (CTS 4.0.1). This change is designated as less restrictive because the Surveillance will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

3.4.1.2

LCO 3.4.5 ~~[~~Two~~]~~ RCS loops shall be OPERABLE and one RCS loop shall be in operation.

----------NOTE----------

3.4.1.2 Footnote *
DOC M03

All reactor coolant pumps may be removed from operation for ≤ 1 hour per 8 hour period, provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1 and

, "SHUTDOWN MARGIN (SDM),"

1

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

Applicability

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.1.2 Action a. A. One RCS loop inoperable.	A.1 Restore RCS loop to OPERABLE status.	72 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~
3.4.1.2 Action a. B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two RCS loops inoperable. OR Required RCS loop not in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND C.2 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately Immediately

DOC M01
3.4.1.2
Action b.

3.4.1.2
Action b.

DOC M02

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify one RCS loop is in operation.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
SR 3.4.5.2	Verify secondary side water level in each steam generator ≥ ~~[25]~~ 10 % narrow range.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.4.1.2.2

4.4.1.2.3

2

2

2

2

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.5.3	----------NOTE---------- Not required to be performed until 24 hours after a required pump is not in operation. ----------	
	Verify correct breaker alignment and indicated power available to each required pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.4.1.2.1

DOC L01

2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

3.4.1.2

LCO 3.4.5 ~~[~~Two~~]~~ RCS loops shall be OPERABLE and one RCS loop shall be in operation.

----------NOTE----------

3.4.1.2 Footnote *

DOC M03

All reactor coolant pumps may be removed from operation for ≤ 1 hour per 8 hour period, provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

1

Applicability

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.1.2 Action a. A. One RCS loop inoperable.	A.1 Restore RCS loop to OPERABLE status.	72 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~
3.4.1.2 Action a. B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two RCS loops inoperable. OR Required RCS loop not in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND C.2 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately Immediately

DOC M01
3.4.1.2
Action b.

3.4.1.2
Action b.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify one RCS loop is in operation.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
SR 3.4.5.2	Verify secondary side water level in each steam generator ≥ ~~[25]~~ 10 % narrow range.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.4.1.2.2

4.4.1.2.3

2

2

2

2

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.5.3	----------NOTE---------- Not required to be performed until 24 hours after a required pump is not in operation. ----------	
	Verify correct breaker alignment and indicated power available to each required pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.4.1.2.1

DOC L01

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.5, RCS LOOPS – MODE 3

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the steam generators (SGs), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP is sufficient to remove core decay heat. However, ~~[~~two~~]~~ RCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE. 2

Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured.

APPLICABLE SAFETY ANALYSES

Analyses have shown that the rod withdrawal event from MODE 3 with one RCS loop in operation is bounded by the rod withdrawal initiated from MODE 2.

Failure to provide heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops - MODE 3 satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require both RCS loops to be available for heat removal, thus providing redundancy. The LCO requires both loops to be OPERABLE with the intent of requiring both SGs to be capable ~~(> 25% water level)~~ of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation. 1

BASES

LCO (continued)

The Note permits a limited period of operation without RCPs. All RCPs may be removed from operation for ≤ 1 hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, a reduction in boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCPs or shutdown cooling (SDC) pump forced circulation (e.g., to change operation from one SDC train to the other, to perform surveillance or startup testing, to perform the transition to and from SDC System cooling, or to avoid operation below the RCP minimum net positive suction head limit). The time period is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE. A RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2,"
LCO 3.4.6, "RCS Loops - MODE 4,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~.

3
3

BASES

ACTIONS

A.1

If one RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the RCS loop to OPERABLE status within a Completion Time of 72 hours ~~[or in accordance with the Risk Informed Completion Time Program]~~. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core. 2

B.1

If restoration for Required Action A.1 is not possible within 72 hours, the unit must be placed in MODE 4 within 12 hours. In MODE 4, the plant may be placed on the SDC System. The Completion Time of 12 hours is compatible with required operation to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If two RCS loops are inoperable the or a required RCS loop is not in operation, except as provided in Note ~~1~~ in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended. Action to restore one RCS loop to OPERABLE status and operation shall be initiated immediately and continued until one RCS loop is restored to OPERABLE status and operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. 1

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification that one RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~[The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

SR 3.4.5.2

narrow range

This SR requires verification that the secondary side water level in each SG is ≥ ~~[25]~~%. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. 2

10

~~[The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within the safety analyses assumptions.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

Verification that each required RCP is OPERABLE ensures that the single failure criterion is met and that an additional RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required RCP. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the steam generators (SGs), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP is sufficient to remove core decay heat. However, ~~[~~two~~]~~ RCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE. 2

Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured.

APPLICABLE SAFETY ANALYSES

Analyses have shown that the rod withdrawal event from MODE 3 with one RCS loop in operation is bounded by the rod withdrawal initiated from MODE 2.

Failure to provide heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops - MODE 3 satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require both RCS loops to be available for heat removal, thus providing redundancy. The LCO requires both loops to be OPERABLE with the intent of requiring both SGs to be capable ~~(> 25% water level)~~ of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation. 1

BASES

LCO (continued)

The Note permits a limited period of operation without RCPs. All RCPs may be removed from operation for ≤ 1 hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, a reduction in boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCPs or shutdown cooling (SDC) pump forced circulation (e.g., to change operation from one SDC train to the other, to perform surveillance or startup testing, to perform the transition to and from SDC System cooling, or to avoid operation below the RCP minimum net positive suction head limit). The time period is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE. A RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2,"
LCO 3.4.6, "RCS Loops - MODE 4,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~.

3
3

BASES

ACTIONS

A.1

If one RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the RCS loop to OPERABLE status within a Completion Time of 72 hours ~~[or in accordance with the Risk Informed Completion Time Program]~~. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core. 2

B.1

If restoration for Required Action A.1 is not possible within 72 hours, the unit must be placed in MODE 4 within 12 hours. In MODE 4, the plant may be placed on the SDC System. The Completion Time of 12 hours is compatible with required operation to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If two RCS loops are inoperable the or a required RCS loop is not in operation, except as provided in Note ~~1~~ in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended. Action to restore one RCS loop to OPERABLE status and operation shall be initiated immediately and continued until one RCS loop is restored to OPERABLE status and operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. 1

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification that one RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~[The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

2

~~]~~

SR 3.4.5.2

narrow range

This SR requires verification that the secondary side water level in each SG is ≥ ~~[25]~~%. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant.

10

2

~~[The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within the safety analyses assumptions.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

2

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.3

Verification that each required RCP is OPERABLE ensures that the single failure criterion is met and that an additional RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required RCP. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES None.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.5, BASES, RCS LOOPS – MODE 3

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the correct Specification Title.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.5, RCS LOOPS – MODE 3

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

3.4.6, RCS Loops – MODE 4

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

consisting of any combination of RCS loops and shutdown cooling (SDC) loops

LCO 3.4.6 ~~3.4.1.3 At least~~ two ~~of the~~ loops ~~listed below~~ shall be OPERABLE and ~~at least~~ one ~~reactor coolant or shutdown cooling loop~~ shall be in operation.*

loop

~~a. Reactor Coolant Loop A and its associated steam generator and at least one associated reactor coolant pump,~~

~~b. Reactor Coolant Loop B and its associated steam generator and at least one associated reactor coolant pump,~~

~~c. Shutdown Cooling Loop A,~~

~~d. Shutdown Cooling Loop B.~~

LA01

Applicability **APPLICABILITY**: MODE 4.

ACTION:

IN One

ACTION A.1 ~~a. With less than the above~~ required ~~reactor coolant or shutdown cooling~~ loop~~s~~ OPERABLE, ~~within one (1) hour~~ initiate corrective action to ~~return the required~~ loop~~s~~ to OPERABLE status and immediately initiate action to make at least one steam generator available for decay heat removal via natural circulation. LCO 3.0.4.a is not applicable when entering ~~HOT~~ ~~SHUTDOWN~~.

immediately M01

restore a second

ACTION A.2

ACTION A.2 Note

MODE 4

Two required loops inoperable OR Required

M02

not

ACTION B.1 ~~b. With no reactor coolant or shutdown cooling~~ loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of ~~Technical~~ ~~Specification~~ 3.1.1.~~1~~ and ~~within one (1) hour~~ initiate ~~corrective~~ action to ~~return the required reactor coolant~~ loop to operation.

immediately M01

LCO

ACTION B.2

immediately

restore one

OPERABLE status and

M01

LCO 3.4.6 Note 1 * All reactor coolant pumps (RCPs) and ~~shutdown cooling~~ SDC pumps may be ~~de-energized~~ removed from operation for ≤ ~~up to~~ 1 hour provided (1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1.~~1~~ and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

M03

per 8 hour period

LCO

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.4.6.3 Note

L01

SR 3.4.6.3 ~~4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE~~ in accordance with the Surveillance Frequency Control Program by verify~~ing~~ correct breaker alignment~~s~~ and indicated power availabi~~lity~~.

le to each required pump

in

SG(s) is

SR 3.4.6.2 ~~4.4.1.3.2 The~~ required ~~steam generator(s) shall be determined OPERABLE by~~ verify~~ing the~~ secondary side water level ~~to be~~ ≥ 10% ~~of narrow range indication~~ in accordance with the Surveillance Frequency Control Program.

LA02

required

RCS loop

SDC

SR 3.4.6.1 ~~4.4.1.3.3 At least one reactor coolant~~ or ~~shutdown cooling~~ loop shall be verified ~~to be~~ in operation ~~and circulating reactor coolant~~ in accordance with the Surveillance Frequency Control Program.

LA02

SDC loop

SR 3.4.6.4 ~~4.4.1.3.4~~ Verify required ~~shutdown cooling train~~ locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.*

SR 3.4.6.4 Note * Not required to be performed until 12 hours after entering MODE 4.

A01

~~REACTOR COOLANT SYSTEM~~

~~REACTOR COOLANT PUMP - STARTING~~

LIMITING CONDITION FOR OPERATION

ITS 3.4.6 Note 2

~~3.4.14~~ If the steam generator temperature exceeds the primary temperature by more than 30°F, the first idle reactor coolant pump shall not be started.

APPLICABILITY: MODES 4[#] and 5.

See ITS 3.4.7

ACTION:

~~If a reactor coolant pump is started when the steam generator temperature exceeds primary temperature by more than 30°F, evaluate the subsequent transient to determine compliance with Specification 3.4.9.1.~~

A02

SURVEILLANCE REQUIREMENTS

ITS 3.4.6 Note 2

~~4.4.14~~ Prior to starting a reactor coolant pump, verify that the steam generator temperature does not exceed primary temperature by more than 30°F.

ITS 3.4.6 Applicability

Any Reactor Coolant System Cold Leg Temperature is ~~less than~~ ≤ 300°F.

A01

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

consisting of any combination of RCS loops and shutdown cooling (SDC) loops

LCO 3.4.6 ~~3.4.1.3 At least~~ two ~~of the~~ loop~~(s)~~/train(s) ~~listed below~~ shall be OPERABLE and ~~at least~~ one ~~Reactor Coolant and/or shutdown cooling loops~~ shall be in operation.*

loop

LA01

~~a. Reactor Coolant Loop 2A and its associated steam generator and at least one associated Reactor Coolant pump,~~**

~~b. Reactor Coolant Loop 2B and its associated steam generator and at least one associated Reactor Coolant pump,~~**

~~c. Shutdown Cooling Train 2A,~~

~~d. Shutdown Cooling Train 2B.~~

LA01

Applicability **APPLICABILITY:** MODE 4.

ACTION:

IN One

ACTION A.1 ~~a. With less than the above~~ required ~~Reactor Coolant and/or shutdown cooling~~ loop~~s~~ OPERABLE, immediately initiate corrective action to

restore a second

~~return the required~~ loop~~s~~ to OPERABLE status ~~as soon as possible~~ and

ACTION A.2 immediately initiate action to make at least one steam generator available for decay heat removal via natural circulation.

ACTION A.2 Note LCO 3.0.4.a is not applicable when entering ~~HOT SHUTDOWN~~.

MODE 4

Two required loops inoperable OR Required

M02

not

ACTION B.1 ~~b. With no Reactor Coolant or shutdown cooling~~ loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of ~~Technical~~

immediately M01

LCO ~~Specifications~~ 3.1.1~~.1~~ and immediately initiate ~~corrective~~ action to ~~return the required coolant~~ loop to operation.

restore one

ACTION B.2

M01

OPERABLE status and

(RCPs) SDC removed from operation

LCO 3.4.6 Note 1 * All Reactor Coolant pumps and ~~shutdown cooling~~ pumps may be ~~de-energized~~

≤ ~~up to~~ 1 hour provided (1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1~~.1~~ and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

M03

per 8 hour period

No RCP RCS any

LCO 3.4.6 Note 2 ** ~~A Reactor Coolant pump~~ shall ~~not~~ be started with two idle loops and ~~one or more of the Reactor Coolant System~~ cold leg temperatures ~~less than or equal to that specified in Table 3.4-3~~ unless ~~the~~ secondary water temperature ~~of~~ each steam generator is less than 40°F above each of the ~~Reactor Coolant System~~ cold leg temperatures.

≤ 252°F

in side

(SG)

RCS

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.4.6.3 Note

L01

SR 3.4.6.3 ~~4.4.1.3.1~~ ~~The required Reactor Coolant pump(s), if not in operation, shall be determined to be OPERABLE~~ in accordance with the Surveillance Frequency Control Program by verify~~ing~~ correct breaker alignment~~s~~ and indicated power availabil~~ity~~.

le to each required pump

in

SG(s) is

SR 3.4.6.2 ~~4.4.1.3.2~~ ~~The~~ required ~~steam generator(s) shall be determined OPERABLE by~~ verify~~ing the~~ secondary side water level ~~to be~~ ≥ 10% ~~indicated narrow range level~~ in accordance with the Surveillance Frequency Control Program.

LA02

required

RCS loop

SDC

SR 3.4.6.1 ~~4.4.1.3.3~~ ~~At least one Reactor Coolant~~ or ~~shutdown cooling~~ loop shall be verified ~~to be~~ in operation ~~and circulating Reactor Coolant~~ in accordance with the Surveillance Frequency Control Program.

LA02

SDC loop

SR 3.4.6.4 ~~4.4.1.3.4~~ Verify required ~~shutdown cooling trains~~ locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.*

SR 3.4.6.4 Note * Not required to be performed until 12 hours after entering MODE 4.

DISCUSSION OF CHANGES
ITS 3.4.6, RCS LOOPS – MODE 4

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 1 only:** CTS 3.4.14 Action states, in part, if a RCP is started when not within the limits, evaluate the subsequent transient to determine compliance with CTS 3.4.9.1, RCS Pressure and Temperature (P/T) Limits. ITS 3.4.6 does not contain a reference to specification 3.4.9.1. (i.e., ITS 3.4.3). This changes the CTS by not including this reference in the ITS.

The purpose of the reference is to alert the user that a determination must be made to ensure compliance with CTS 3.4.9.1, RCS P/T Limits. It is an ITS convention to not include these types of cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.4.1.3 Action a. states, in part, with less than the above required reactor coolant or shutdown cooling loops OPERABLE, within one (1) hour initiate corrective action to return the required loops to OPERABLE status. ITS 3.4.6 ACTION A states that with one required loop inoperable, “immediately” initiate action to restore a second loop to OPERABLE status. This changes the Unit 1 CTS by adding the requirement to “immediately” initiate action to restore a second loop to OPERABLE status.”

Unit 1 and Unit 2 CTS 3.4.1.3 Action b. states, in part, with no Reactor Coolant or shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specifications 3.1.1.1. ITS 3.4.6 ACTION B states that with two required loops inoperable or required loop not in operation, “immediately” suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. This changes the CTS by adding the requirement to “immediately” suspend operations that would cause introduction into the RCS. | R1 | R1

Unit 1 CTS 3.4.1.3 Action b. states, in part, and within one (1) hour initiate corrective action to return the required reactor coolant loop to operation. Unit 2 CTS Action b. states, in part, and immediately initiate corrective action to return the required coolant loop to operation. ITS 3.4.6 ACTION B states that with two required loops inoperable or required loop not in operation, immediately initiate action to restore one loop to OPERABLE status and operation. This changes the

Unit 1 CTS by adding the requirement to "immediately" initiate action to restore one loop to OPERABLE status and operation and changes the Unit 2 CTS and by adding the requirement to "immediately" initiate action to restore one loop to OPERABLE status and operation. | R1

This change is acceptable because it provides appropriate actions if one required loop is inoperable, and if two required loops or trains are inoperable or the required loop is not in operation. Under these conditions immediate action is necessary to ensure unit transients do not occur, and action is immediately taken to restore one loop to OPERABLE status and operation to be able to remove the decay heat generated by the reactor. These actions are required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operation. Action to restore one RCS loop to OPERABLE status and operation is necessary to be able to remove the decay heat generated by the reactor. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. This change is designated as more restrictive, because it requires immediate action to return the required reactor coolant loop to operation and adds the requirement to restore the loop to OPERABLE status.

M02 CTS 3.4.1.3 Action b. states, in part, that with no reactor coolant loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specification 3.1.1.1. ITS 3.4.6 Action B states the Condition two required loops inoperable or required loop not in operation. This changes the CTS by adding the Condition "two required loops inoperable" to the CTS Condition no required loop in operation.

This change is acceptable because it provides appropriate actions if two required loops are inoperable or a required loop is not in operation. Under these conditions, immediate action is necessary to ensure certain unit transients do not occur, and action is immediately taken to restore one loop to OPERABLE status and operation to be able to remove the decay heat generated by the reactor. This change is designated as more restrictive because it requires an additional condition under which the actions must be taken.

M03 CTS 3.4.1.3 requires, in part, that at least two of the loops listed shall be OPERABLE and at least one shall be in operation. CTS 3.4.1.3 is modified by a footnote (*) that states, in part, that all reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour under the conditions specified therein. ITS 3.4.6 contains the same allowance but limits the use of the 1 hour exception to once per 8 hour period. This changes the CTS by modifying the 1 hour allowance that all reactor coolant pumps and shutdown cooling (SDC) pumps may be removed from operation under the conditions specified therein and limits the usage of the allowance to once per 8 hour period. | R1

The purpose of the 1 hour allowance is to allow reactor coolant pumps and SDC pumps to be removed from operation to support testing. This means that natural circulation has been established using the SGs. This change is acceptable because it helps ensure that boron stratification and inadequate decay heat

removal do not occur should multiple 1 hour periods be required. This change is designated as more restrictive because it limits the allowance to 1 hour per 8 hour period, and that restriction does not currently exist

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.4.1.3 states, in part, that at least two of the loops listed shall be OPERABLE and at least one reactor coolant or shutdown cooling loop shall be in operation. Unit 2 CTS 3.4.1.3 states, in part, that at least two of the loop(s)/train(s) listed shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation. CTS 3.4.1.3 contains a description of what constitutes OPERABLE RCS loops and SDC loops. ITS 3.4.5 does not include a description of what constitutes OPERABLE RCS loops and SDC loops. This changes the CTS by moving the details of what constitutes OPERABLE RCS loops and SDC loops to the Bases. Additionally, Unit 2 CTS 3.4.1.3 is also changed by deletion of the descriptor "train(s)". Use of the descriptor "SDC loop(s)" is consistent with Unit 1 CTS 3.4.1.3, CTS 3.4.1.4.1, "Cold Shutdown – Loops Filled," CTS 3.4.1.4.2, "Cold Shutdown – Loops Not Filled," CTS 3.9.8.1, "Shutdown Cooling and Coolant Circulation," and CTS 3.9.8.2, "Low Water Level."

The removal of these details related to system design from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains a requirement for two loops consisting of any combination of RCS and SDC loops to be OPERABLE. The removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.3.2 states, in part, that the required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be > 10% of narrow range indication. ITS 3.4.6.2 requires verifying secondary side water level in required SG(s) is > 10%. This changes the CTS by moving the Surveillance Requirement for the instrumentation to be used to verify secondary side level in required SGs, to the Bases. CTS 4.4.1.3.3 states, in part, that at least one reactor coolant or shutdown cooling loop shall be verified to be in operation and circulating reactor coolant. ITS SR 3.4.6.1 states verify required RCS loop or SDC loop is in operation. This changes the CTS by moving the Surveillance Requirement to verify that the reactor coolant loops or SDC loops are circulating reactor coolant, to the Bases.

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that two loops consisting of any combination of RCS loops and SDC loops shall be OPERABLE and one loop shall be in operation. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal. Also, this change is acceptable because this type of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 (Category 7 – Relaxation of Surveillance Frequency) CTS 4.4.1.3.1 states, in part, that the required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability. ITS SR 3.4.6.3 requires verification of correct breaker alignment and indicated power availability to the pump not in operation in accordance with the Surveillance Frequency Control Program. It is modified by a Note that states "Not required to be performed until 24 hours after a required pump is not in operation." This changes the CTS by not requiring the Surveillance Requirement to be performed until 24 hours after a pump is taken out of operation.

The purpose of CTS 4.4.1.3.1 is to ensure that the standby pump is ready to operate. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The Note provides time to perform the Surveillance to verify correct breaker alignment and indicated power availability. Without the Note, the Surveillance would not be met immediately after taking a pump out of operation. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

3.4.1.3

LCO 3.4.6 Two loops ~~or trains~~ consisting of any combination of RCS loops and shutdown cooling (SDC) ~~trains~~ loops shall be OPERABLE and one loop ~~or train~~ shall be in operation. 1

----------NOTES----------

3.4.1.3 Footnote *

DOC M03

1. All reactor coolant pumps (RCPs) and SDC pumps may be removed from operation for ≤ 1 hour per 8 hour period, provided:

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," and 1

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

3.4.14

2. No RCP shall be started with two idle RCS loops and any RCS cold leg temperature ≤ 300°F unless ~~less than or equal to the LTOP enable temperature specified in the PTLR unless:~~ 1

   ~~a. Pressurizer water level is < [60]% or~~

   ~~b.~~ Secondary side water temperature in each steam generator (SG) is < ~~[100]~~ 30°F above each of the RCS cold leg temperatures.

Applicability **APPLICABILITY:** MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.1.3 Action a. A. One required loop inoperable.	A.1 Initiate action to restore a second loop ~~or train~~ to OPERABLE status. 1 AND	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
3.4.1.3 Action a.	A.2	----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Initiate action to make at least one steam generator available for decay heat removal via natural circulation.	Immediately
DOC M02 3.4.1.3 Action b. B. Two required loops ~~or trains~~ inoperable. <u>OR</u> Required loop ~~or train~~ not in operation.	B.1	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately (1) (1)
	<u>AND</u>		
3.4.1.3 Action b.	B.2	Initiate action to restore one loop ~~or train~~ to OPERABLE status and operation.	Immediately (1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.4.1.3.3 SR 3.4.6.1	Verify required RCS loop or SDC ~~train~~ loop is in operation.	~~[12 hours~~ (1) (2) ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ (2)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.1.3.2	SR 3.4.6.2	Verify secondary side water level in required SG(s) is ≥ ~~[25]~~%. (10) (narrow range)	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.1.3.1 DOC L01	SR 3. 4.6.3	----------NOTE---------- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to each required pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.1.3.4	SR 3.4. 6.4	----------NOTE---------- Not required to be performed until 12 hours after entering MODE 4. ---------- Verify required SDC ~~train~~ (loop) locations susceptible to gas accumulation are sufficiently filled with water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 1 2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

3.4.1.3

LCO 3.4.6

Two loops ~~or trains~~ consisting of any combination of RCS loops and shutdown cooling (SDC) ~~trains~~ loops shall be OPERABLE and one loop ~~or train~~ shall be in operation. 1

----------NOTES----------

3.4.1.3 Footnote *

DOC M03

1. All reactor coolant pumps (RCPs) and SDC pumps may be removed from operation for ≤ 1 hour per 8 hour period, provided:

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," and 1

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

3.4.1.3 Footnote **

2. No RCP shall be started with any RCS cold leg temperature two idle RCS loops and ~~less than or equal to the LTOP enable temperature specified in the PTLR unless:~~ ≤ 252°F unless 1

   ~~a. Pressurizer water level is < [60]% or~~

   ~~b.~~ Secondary side water temperature in each steam generator (SG) is < ~~[100]~~ 40°F above each of the RCS cold leg temperatures.

Applicability

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.1.3 Action a. A. One required loop inoperable.	A.1 Initiate action to restore a second loop ~~or train~~ to OPERABLE status. 1 AND	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.1.3 Action a.	A.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Initiate action to make at least one steam generator available for decay heat removal via natural circulation.	Immediately
DOC M02 3.4.1.3 Action b. B. Two required loops ~~or trains~~ inoperable. OR Required loop ~~or train~~ not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND	Immediately (1) (1)
3.4.1.3 Action b.	B.2 Initiate action to restore one loop ~~or train~~ to OPERABLE status and operation.	Immediately (1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.4.1.3.3 SR 3.4.6.1	Verify required RCS loop or SDC ~~train~~ loop is in operation.	~~[12 hours~~ (1) (2) ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ (2)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.1.3.2	SR 3.4.6.2	Verify secondary side water level in required SG(s) is ≥ ~~[25]~~%. (10) (narrow range)	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.1.3.1 DOC L01	SR 3. 4.6.3	----------NOTE---------- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to each required pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.1.3.4	SR 3.4. 6.4	----------NOTE---------- Not required to be performed until 12 hours after entering MODE 4. ---------- Verify required SDC ~~train~~ (loop) locations susceptible to gas accumulation are sufficiently filled with water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.6, RCS LOOPS – MODE 4

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the steam generators (SGs) or shutdown cooling (SDC) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either Reactor Coolant ~~pumps (RCPs)~~ or SDC ~~trains~~ can be used for coolant circulation. The intent of this LCO is to provide forced flow from at least one ~~RCP~~ or one SDC ~~train~~ for decay heat removal and transport. The flow provided by one RCP loop or SDC ~~train~~ is adequate for heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for heat removal. 1

System (RCS) loops; loops; reactor coolant pump (RCP); pump; loop

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS loops and SDC ~~trains~~ provide this circulation. 1

loops

RCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two loops ~~or trains~~, RCS or SDC, be OPERABLE in MODE 4 and one of these loops ~~or trains~~ be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS and SDC System loops. Any one loop ~~or train~~ in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop ~~or train~~ is required to be OPERABLE to provide redundancy for heat removal. 1

Note 1 permits all RCPs and SDC pumps to be removed from operation ≤ 1 hour per 8 hour period. This means that natural circulation has been established using the SGs. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the RCS without the RCPs or SDC pumps depends on the core decay heat load and the length of time that the

BASES

LCO (continued)

pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) limits or low temperature overpressure protection (LTOP) limits) must be observed and forced SDC flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both RCPs or SDC pumps are to be limited to situations where:

a. Pressure and temperature increases can be maintained well within the allowable pressure (P/T limits and LTOP) and 10°F subcooling limits or

b. An alternate heat removal path through the SGs is in operation.

Note 2 requires that ~~either of the following two conditions be satisfied~~ before an RCP may be started with any RCS cold leg temperature ~~less than or equal to the LTOP enable temperature specified in the PTLR:~~

two idle RCS loops and

1

≤ 300°F,

~~a. Pressurizer water level is < [60]% or~~

b. Secondary side water temperature in each SG is < [100]°F above each of the RCS cold leg temperatures.

30

1

Satisfying ~~either of~~ the above condition~~s~~ will preclude a large pressure surge in the RCS when the RCP is started.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE and has the minimum water level specified in SR 3.4.6.2.

loop

Similarly, for the SDC System, an OPERABLE SDC ~~train~~ is composed of the OPERABLE SDC pump~~(s)~~ capable of providing forced flow to the SDC heat exchanger~~(s)~~. RCPs and SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. Management of gas voids is important to SDC System OPERABILITY.

1

BASES

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs or the SDC System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2,"
LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and
LCO 3.9.5, "Shutdown Cooling and Coolant Circulation - Low Water Level" ~~(MODE 6)~~.

3
3

ACTIONS

A.1

If only one required ~~RCS~~ loop is OPERABLE and in operation ~~and no SDC trains are OPERABLE~~, redundancy for heat removal is lost. Action must be initiated immediately to restore a required non-operating loop ~~or train~~ to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for decay heat removal.

1

A.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

4

loop

If only one required ~~SDC train~~ is OPERABLE and in operation ~~and no required RCS loops are OPERABLE~~, redundancy for heat removal is lost and the plant must be placed in a configuration that minimizes overall plant risk. This redundancy is obtained by making at least one SG available for decay heat removal via natural circulation because:

1

1

BASES

ACTIONS (continued)

1. MODE 4 operation poses overall lower risk of core damage and large early radiation release than does MODE 5 (Ref. 1). This is particularly true with SDC impaired.

2. In MODE 4, RCS and steam generator loop conditions may be maintained such that failure of the operating SDC ~~train~~ may be by natural circulation heat removal through one or more steam generators. 1

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 1). However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action A.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

B.1 and B.2

If two required loops ~~or trains~~ are inoperable or a required loop ~~or train~~ is not in operation except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one ~~RCS~~ loop ~~or SDC train~~ to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of decay heat removal. The action to restore must continue until one loop ~~or train~~ is restored to operation. 1 1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

RCS loop or SDC

This SR requires verification that the required loop ~~or train~~ is in operation. This ensures forced flow is providing heat removal. Verification includes flow rate, temperature, or pump status monitoring. ~~[The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.4.6.2

narrow range

This SR requires verification of secondary side water level in the required SG(s) ≥ ~~[25]~~%. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. ~~[The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.~~

10

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.3

loop

Verification that each required pump is OPERABLE ensures that an additional RCS loop or SDC ~~train~~ can be placed in operation, if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

2

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

SR 3.4.6.4

SDC System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the required SDC ~~train~~(s) and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

loop

1

Selection of SDC System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

The SDC System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the SDC System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

SDC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

This SR is modified by a Note that states the SR is not required to be performed until 12 hours after entering MODE 4. In a rapid shutdown, there may be insufficient time to verify all susceptible locations prior to entering MODE 4.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the SDC System piping and the procedural controls governing system operation.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

REFERENCES

1. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the steam generators (SGs) or shutdown cooling (SDC) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

Reactor Coolant System (RCS) loops | loops | R1

In MODE 4, either ~~reactor coolant pumps (RCPs)~~ or SDC ~~trains~~ can be used for coolant circulation. The intent of this LCO is to provide forced flow from at least one ~~RCP~~ or one SDC ~~train~~ for decay heat removal and transport. The flow provided by one RCP loop or SDC ~~train~~ is adequate for heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for heat removal. | 1

reactor coolant pump (RCP) | pump | loop

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS loops and SDC ~~trains~~ provide this circulation. | 1

loops

RCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two loops ~~or trains~~, RCS or SDC, be OPERABLE in MODE 4 and one of these loops ~~or trains~~ be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS and SDC System loops. Any one loop ~~or train~~ in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop ~~or train~~ is required to be OPERABLE to provide redundancy for heat removal. | 1

Note 1 permits all RCPs and SDC pumps to be removed from operation ≤ 1 hour per 8 hour period. This means that natural circulation has been established using the SGs. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. The response of the RCS without the RCPs or SDC pumps depends on the core decay heat load and the length of time that the

BASES

LCO (continued)

pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) limits or low temperature overpressure protection (LTOP) limits) must be observed and forced SDC flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both RCPs or SDC pumps are to be limited to situations where:

a. Pressure and temperature increases can be maintained well within the allowable pressure (P/T limits and LTOP) and 10°F subcooling limits or

b. An alternate heat removal path through the SGs is in operation.

Note 2 requires that ~~either of the following two conditions be satisfied~~ before an RCP may be started with any RCS cold leg temperature ~~less than or equal to the LTOP enable temperature specified in the PTLR:~~ ≤ 252°F,

two idle RCS loops and

1

~~a. Pressurizer water level is < [60]% or~~

b. Secondary side water temperature in each SG is < [100]°F above each of the RCS cold leg temperatures.

40

1

Satisfying ~~either of~~ the above condition~~s~~ will preclude a large pressure surge in the RCS when the RCP is started.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE and has the minimum water level specified in SR 3.4.6.2.

loop

Similarly, for the SDC System, an OPERABLE SDC ~~train~~ is composed of the OPERABLE SDC pump~~(s)~~ capable of providing forced flow to the SDC heat exchanger~~(s)~~. RCPs and SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. Management of gas voids is important to SDC System OPERABILITY.

1

BASES

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs or the SDC System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2,"
LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and
LCO 3.9.5, "Shutdown Cooling and Coolant Circulation - Low Water Level" ~~(MODE 6)~~.

3
3

ACTIONS

A.1

If only one required ~~RCS~~ loop is OPERABLE and in operation ~~and no SDC trains are OPERABLE~~, redundancy for heat removal is lost. Action must be initiated immediately to restore a required non-operating loop ~~or train~~ to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for decay heat removal.

1

A.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

4

loop

If only one required ~~SDC train~~ is OPERABLE and in operation ~~and no required RCS loops are OPERABLE~~, redundancy for heat removal is lost and the plant must be placed in a configuration that minimizes overall plant risk. This redundancy is obtained by making at least one SG available for decay heat removal via natural circulation because:

1

1

BASES

ACTIONS (continued)

1. MODE 4 operation poses overall lower risk of core damage and large early radiation release than does MODE 5 (Ref. 1). This is particularly true with SDC impaired.

2. In MODE 4, RCS and steam generator loop conditions may be maintained such that failure of the operating SDC ~~train~~ may be mitigated by natural circulation heat removal through one or more steam generators. 1

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 1). However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action A.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

B.1 and B.2

If two required loops ~~or trains~~ are inoperable or a required loop ~~or train~~ is not in operation except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one ~~RCS~~ loop ~~or SDC train~~ to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of decay heat removal. The action to restore must continue until one loop ~~or train~~ is restored to operation. 1 1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

RCS loop or SDC

This SR requires verification that the required loop ~~or train~~ is in operation. This ensures forced flow is providing heat removal. Verification includes flow rate, temperature, or pump status monitoring. ~~[The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.4.6.2

narrow range

This SR requires verification of secondary side water level in the required SG(s) ≥ ~~[25]~~%. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. ~~[The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.~~

10

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.3

loop

Verification that each required pump is OPERABLE ensures that an additional RCS loop or SDC ~~train~~ can be placed in operation, if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

2

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

SR 3.4.6.4

SDC System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the required SDC ~~train~~(s) and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

loop

1

Selection of SDC System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The SDC System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the SDC System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

SDC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

This SR is modified by a Note that states the SR is not required to be performed until 12 hours after entering MODE 4. In a rapid shutdown, there may be insufficient time to verify all susceptible locations prior to entering MODE 4.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the SDC System piping and the procedural controls governing system operation.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

REFERENCES

1. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.6, BASES, RCS LOOPS – MODE 4

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the correct Specification Title.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.6, RCS LOOPS – MODE 4

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

3.4.7, RCS Loops – MODE 5, Loops Filled

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTOR COOLANT SYSTEM

COLD SHUTDOWN – LOOPS FILLED

LIMITING CONDITION FOR OPERATION

LCO 3.4.7

~~3.4.1.4.1~~ ~~At least~~ one shutdown cooling loop shall be OPERABLE and in operation* and either:

~~a.~~ One additional shutdown cooling loop shall be OPERABLE#, or

~~b.~~ The secondary side water level of ~~at least two~~ steam generator~~s~~ shall be ~~greater than~~ 10% ~~of narrow range indication~~.

each

≥

RCS

R1

L02

Applicability

APPLICABILITY: MODE 5 with ~~reactor coolant~~ loops filled##.

ACTION:

s with secondary side water level not within limit

AND One SDC loop OPERABLE

A02

One

SDC

IN

Condition A

Condition B

ACTION A,B

One or more

restore a second SDC

Immediately initiate action

~~a.~~ ~~With less than the above~~ required loop~~s~~ OPERABLE or ~~with less than the~~ required steam generator ~~level~~, ~~within one (1) hour~~ initiate ~~corrective~~ action to ~~return the required~~ loop~~s~~ to OPERABLE status or to restore ~~the~~ required ~~level~~.

immediately

M01

SGs secondary side water level to within limit

No required SDC loops OPERABLE OR Required SDC

not

immediately

M01

ACTION C

~~b.~~ ~~With no shutdown cooling~~ loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1~~.2~~ and ~~within one (1) hour~~ initiate ~~corrective~~ action to ~~return the~~ required shutdown loop to operation.

restore one

LCO

immediately

M01

OPERABLE status and

M01

SURVEILLANCE REQUIREMENTS

SR 3.4.7.2

~~4.4.1.4.1.1~~ ~~The~~ secondary side water level ~~of at least two steam generators when required shall be determined to be within limits~~ in accordance with the Surveillance Frequency Control Program.

Verify required SG

is ≥ 10%

is

SR 3.4.7.1

~~4.4.1.4.1.2~~ ~~At least one shutdown cooling~~ loop ~~shall be determined to be~~ in operation ~~and circulating reactor coolant~~ in accordance with the Surveillance Frequency Control Program.

Verify required SDC

LA01

Insert proposed SR 3.4.7.3

M02

SR 3.4.7.4

~~4.4.1.4.1.3~~ Verify required ~~shutdown cooling train~~ locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

SDC loop

SDC

of the loop in operation

removed from operation

≤

per 8 hour period

M03

LCO 3.4.7 Note 1

* The ~~shutdown cooling~~ pump may be ~~de-energized~~ for ~~up to~~ 1 hour provided 1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1~~.2~~ and 2) core outlet temperature is maintained ~~at least~~ 10°F below saturation temperature.

SDC

≥

LCO

R1

LCO 3.4.7 Note 2

\# One ~~shutdown cooling~~ loop may be inoperable for up to 2 hours for surveillance testing provided the other ~~shutdown cooling~~ loop is OPERABLE and in operation.

No RCP

SDC

LCO 3.4.7 Note 3

~~A reactor coolant pump~~ shall ~~not~~ be started with two idle loops unless ~~the~~ secondary water temperature ~~of~~ each steam generator is less than 30°F above each of the ~~Reactor Coolant System~~ cold leg temperatures.

side

RCS

in

Insert proposed LCO 3.4.7 Note 4

L01

~~REACTOR COOLANT SYSTEM~~

~~REACTOR COOLANT PUMP - STARTING~~

LIMITING CONDITION FOR OPERATION

ITS 3.4.7 Note 3

~~3.4.14~~ If the steam generator temperature exceeds the primary temperature by more than 30°F, the first idle reactor coolant pump shall not be started.

APPLICABILITY: MODES 4[#] and 5.

ACTION:

If a reactor coolant pump is started when the steam generator temperature exceeds primary temperature by more than 30°F, evaluate the subsequent transient to determine compliance with Specification 3.4.9.1.

See ITS 3.4.6

SURVEILLANCE REQUIREMENTS

ITS 3.4.7 Note 3

~~4.4.14~~ Prior to starting a reactor coolant pump, verify that the steam generator temperature does not exceed primary temperature by more than 30°F.

Reactor Coolant System Cold Leg Temperature is less than 300°F.

See ITS 3.4.6

REACTOR COOLANT SYSTEM

COLD SHUTDOWN – LOOPS FILLED

CONDITION FOR OPERATION

LCO 3.4.7

~~3.4.1.4.1~~ ~~At least~~ one shutdown cooling loop shall be OPERABLE and in operation*, and either:

~~a.~~ One additional shutdown cooling loop shall be OPERABLE#, or

~~b.~~ The secondary side water level of ~~at least two~~ steam generator~~s~~ shall be greater than 10% ~~indicated narrow range level~~.

each | ≥ | RCS | R1

Applicability

APPLICABILITY: MODE 5 with ~~Reactor Coolant~~ loops filled##.

L02 | s with secondary side water level not within limit | AND One SDC loop OPERABLE | A02

ACTION:

required | M04

Condition A
Condition B
ACTION A,B

~~a.~~ ~~With~~ one ~~of the~~ shutdown cooling loop~~s~~ inoperable ~~and with less than the~~ required steam generator ~~level~~, immediately initiate ~~corrective~~ action to ~~return the inoperable shutdown cooling~~ loop to OPERABLE status or to restore ~~the~~ required steam generator ~~level as soon as possible~~.

One or more | restore a second SDC | Immediately initiate action

s secondary side water level to within limit | immediately | M01

ACTION C

No required SDC loops OPERABLE OR Required SDC | not

~~b.~~ ~~With no shutdown cooling~~ loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1~~.2~~ and immediately initiate ~~corrective~~ action to ~~return the~~ required shutdown cooling loop to operation.

restore one | LCO | OPERABLE status and | M01

SURVEILLANCE REQUIREMENTS

SR 3.4.7.2

~~4.4.1.4.1.1~~ ~~The~~ secondary side water level ~~of at least two steam generators when required shall be determined to be within limits~~ in accordance with the Surveillance Frequency Control Program.

Verify required SG | is ≥ 10%

SR 3.4.7.1

~~4.4.1.4.1.2~~ ~~At least one shutdown cooling~~ loop ~~shall be determined to be~~ in operation ~~and circulating reactor coolant~~ in accordance with the Surveillance Frequency Control Program.

is | Verify required SDC | LA01

Insert proposed SR 3.4.7.3 | M02

SR 3.4.7.4

~~4.4.1.4.1.3~~ Verify required ~~shutdown cooling trains~~ locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

SDC loop

SDC | of the loop in operation | removed from operation | ≤ | per 8 hour period | M03

LCO 3.4.7 Note 1

* The ~~shutdown cooling~~ pump may be ~~de-energized~~ for ~~up to~~ 1 hour provided 1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1~~.2~~ and 2) core outlet temperature is maintained ~~at least~~ 10°F below saturation temperature.

LCO | ≥ | R1

SDC

LCO 3.4.7 Note 2

\# One ~~shutdown cooling~~ loop may be inoperable for up to 2 hours for surveillance testing provided the other ~~shutdown cooling~~ loop is OPERABLE and in operation.

SDC

No RCP

LCO 3.4.7 Note 3

~~A Reactor Coolant pump~~ shall ~~not~~ be started with two idle loops unless ~~the~~ secondary water temperature ~~of~~ each steam generator is less than 40°F above each of the ~~Reactor Coolant System~~ cold leg temperatures.

side | RCS | in

Insert proposed LCO 3.4.7 Note 4 | L01

DISCUSSION OF CHANGES
ITS 3.4.7, RCS LOOPS – MODE 5, LOOPS FILLED

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 ITS SR 3.4.7.3 Condition A requires one required shutdown cooling (SDC) loop inoperable and “one SDC loop OPERABLE.” ITS SR 3.4.7.3 Condition B requires one or more SGs with secondary water level not within limit and “one SDC loop OPERABLE.” The condition for “one SDC loop OPERABLE” is not in the CTS. This changes the CTS by adding an additional condition; however, this condition requires one “SDC loop be OPERABLE” which is expected if one SDC loop is inoperable, or if one or more SGs with secondary water level not within limit. The addition of this condition to ITS Condition A and Condition B allows entry into ITS 3.4.7 ACTION C, that is, no required SDC loops OPERABLE or required SDC loop not in operation, without requiring entry into ACTION A or ACTION B. These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.4.1.4.1 Action a. states, in part, with less than the above required loops OPERABLE or with less than the required steam generator level, within one (1) hour initiate corrective action to return the required loops to OPERABLE status or to restore the required level. ITS 3.4.7 Required Actions A.1 and B.1 each state “immediately” initiate action to restore a second SDC loop to OPERABLE status. ITS 3.4.7 Required Actions A.2 and B.2 each state “immediately” initiate action to restore required SGs secondary side water level to within limit. This changes the Unit 1 CTS by adding the requirement to “immediately” initiate the action to restore a second loop to OPERABLE status and to restore required SGs secondary side water level to within limit.

CTS 3.4.1.4.1 Action b. states, in part, with no shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specifications 3.1.1.1. ITS 3.4.7 ACTION C and Required Action C. 1 state that with no required SDC loop OPERABLE or required SDC loop is not in operation, “immediately” suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. This changes the Unit 1 and Unit 2 CTS by adding the requirement to “immediately” suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1 to the CTS.

CTS 3.4.1.4.1 Action b. states, in part, with no shutdown cooling loop in operation, within one (1) hour (Unit 1) and immediately (Unit 2) initiate corrective action to return the required shutdown loop to operation. ITS 3.4.7 ACTION C and Required Action C. 2 state that with no required SDC loop OPERABLE or required SDC loop is not in operation, immediately initiate action to restore one SDC loop to OPERABLE status and operation. This changes the CTS by adding the requirement to restore the loop to OPERABLE status.

This change is acceptable because it provides appropriate actions if one SDC loop is inoperable, no required SDC loops are OPERABLE, or a required SDC loop in not is not in operation. Under these conditions immediate action is necessary to ensure unit transients do not occur, and action is immediately taken to restore one loop to OPERABLE status and operation to be able to remove the decay heat generated by the reactor. These actions are required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operation. Action to restore one SDC loop to OPERABLE status and operation is necessary to be able to remove the decay heat generated by the reactor. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. This change is designated as more restrictive, because it requires immediate action to return the required reactor coolant loop to operation and adds the requirement to restore the loop to OPERABLE status.

M02 ITS SR 3.4.7.3 requires verification that correct breaker alignment and indicated power are available to each required SDC pump. A Note further explains that the Surveillance is not required to be performed until 24 hours after a required pump is not in operation. This Surveillance is not required by the CTS. This changes the CTS by requiring verification of correct breaker alignment and indicated power availability on required SDC pumps that are not in operation.

The purpose of ITS SR 3.4.7.3 is to ensure a standby SDC pump is available to provide RCS cooling should the operating pump fail. This change is acceptable because the verification of proper breaker alignment and power availability ensures that an additional SDC pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. This change is designated as more restrictive because it adds a Surveillance Requirement to the CTS.

M03 CTS 3.4.1.4.1 states, in part, that at least one shutdown cooling loop shall be OPERABLE and in operation. CTS 3.4.1.4.1 is modified by a footnote (*) that states, in part, that the shutdown cooling pumps may be de-energized for up to 1 hour under the conditions specified therein. ITS 3.4.7 contains the same allowance but limits the use of the 1 hour exception to once per 8 hour period. This changes the CTS by modifying the 1 hour allowance that SDC pumps may be removed from operation under the conditions specified therein and limits the usage of the allowance to once per 8 hour period.

The purpose of the 1 hour allowance is to allow SDC pumps to be removed from operation to change operation from one SDC loop to the other, perform surveillance or startup testing, and perform the transition to and from the SDC.

This change is acceptable because it helps ensure that boron stratification and inadequate decay heat removal do not occur should multiple 1 hour periods be required. This change is designated as more restrictive because it limits the allowance to 1 hour per 8 hour period, and that restriction does not currently exist.

M04 **Unit 2 only:** CTS 3.4.1.4.1 Action a. states, in part, with one of the shutdown cooling loops inoperable "and" with less than the required steam generator level, immediately initiate the actions specified therein. ITS 3.4.7 ACTION A states that with one required SDC loop inoperable and one SDC loop OPERABLE, then immediately initiate the actions specified therein, of which, are the same as the CTS actions. Similarly, ITS 3.4.7 ACTION B states that with one or more required SGs with secondary side water level not within limit and one SDC loop OPERABLE, then immediately initiate the actions specified therein, of which, are the same as the CTS actions. This changes the CTS by requiring the actions specified therein be taken with one of the shutdown cooling loops inoperable "or" with less than the required steam generator level, rather than both Conditions being required before actions are taken (i.e., and).

This change is acceptable because the ITS retains the appropriate actions if one of the shutdown cooling loops inoperable or if steam generator level is less than the required steam generator level. Under these conditions, immediate action is necessary to ensure certain unit transients do not occur, and action is immediately taken to restore one loop to OPERABLE status and operation to be able to remove the decay heat generated by the reactor. This change is designated as more restrictive, because it adds separate Conditions than are required in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.4.1.2 states, in part, that at least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant. ITS SR 3.4.7.1 states verify required SDC loop is in operation. This changes the CTS by moving the requirement to verify that the SDC loops are circulating reactor coolant, to the Bases. R1

The removal of this detail for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. This change is acceptable because this type of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated

as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.4.1.4.1 states, in part, that at least one shutdown cooling loop shall be OPERABLE and in operation, and either of the specified requirements therein. ITS 3.4.7 specifies the same requirements; however, ITS LCO 3.4.7 Note 4 allows all SDC loops to be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation. This changes the CTS by adding an allowance for all SDC loops to be removed from operation during planned heatup operations to MODE 4.

The purpose of CTS 3.4.1.4.1 is to ensure there is sufficient forced circulation to prevent boric acid stratification and to provide forced flow for decay heat removal and transport. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. This change allows an RCS loop to be in operation instead of a SDC loop. The RCS loop simply replaces the function of the SDC loop. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L02 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.4.1.4.1.b states, in part, that steam generator water level "shall be greater than 10%." ITS LCO 3.4.7.b states "The secondary side water level of each steam generator (SG) shall be ≥ 10%." This changes the CTS by setting the lower SG level limit at a specific value of 10% instead of "greater than" 10%. R1

The purpose of the secondary side steam generator minimum water level is to provide an alternative method of decay heat removal if a second SDC loop is not OPERABLE. The minimum secondary side SG level of ≥10% ensures an alternate method of decay heat removal is available via natural circulation. The value of ≥ 10% is consistent with CTS secondary side water level requirements for HOT STANDBY (MODE 3) and HOT SHUTDOWN (MODE 4) which are retained in ITS 3.4.5 and ITS 3.4.6, respectively. This change is acceptable because a secondary side SG level at ≥ 10% is sufficient to provide for natural circulation decay heat removal. This change is designated as less restrictive because less stringent requirements for OPERABILITY are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

3.4.1.4.1 LCO 3.4.7

One shutdown cooling (SDC) ~~train~~ loop shall be OPERABLE and in operation and either: (1)

a. One additional SDC ~~train~~ loop shall be OPERABLE or (1)

b. The secondary side water level of each steam generator (SG) shall be ≥ ~~[25~~ 10 narrow range %~~]~~. (2) | R1

DOC L02

---NOTES--- loop

3.4.1.4.1 Note *

1. The SDC pump of the ~~train~~ loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided: (1)

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1 , "SHUTDOWN MARGIN (SDM)," and (1)

   b. Core outlet temperature is maintained at ≥ 10°F below saturation temperature.

3.4.1.4.1 Note #

2. One SDC ~~train~~ loop may be inoperable for up to 2 hours for surveillance testing provided that the other SDC ~~train~~ loop is OPERABLE and in operation. (1)

3.4.1.4.1 Note ##

3. No reactor coolant pump (RCP) shall be started with ~~any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR unless:~~ two idle RCS loops unless (1)

   ~~a. The pressurizer water level is < [60]% or~~

   ~~b.~~ ~~The~~ secondary side water temperature in each SG is < ~~[100]~~ 30°F above each of the RCS cold leg temperatures. (2)

DOC L01

4. All SDC ~~trains~~ loops may not be in operation during planned heatup to MODE 4 when at least one RCS loop is in operation. (1)

Applicability APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required SDC ~~train~~ loop inoperable. AND One SDC ~~train~~ loop OPERABLE.	A.1 Initiate action to restore a second SDC ~~train~~ loop to OPERABLE status. OR A.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately Immediately
B. One or more required SGs with secondary side water level not within limit. AND One SDC ~~train~~ loop OPERABLE.	B.1 Initiate action to restore a second SDC ~~train~~ loop to OPERABLE status. OR B.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately Immediately
C. No required SDC ~~trains~~ loops OPERABLE. OR Required SDC ~~train~~ loop not in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND C.2 Initiate action to restore one SDC ~~train~~ loop to OPERABLE status and operation.	Immediately Immediately

3.4.1.4.1 Action A, B
DOC M01
DOC A02
3.4.1.4.1 Action A, B
3.4.1.4.1 Action A, B
DOC A02
3.4.1.4.1 Action C
DOC M01

1 1 1 1 1 1 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.1.4.1.2	SR 3.4.7.1	Verify required SDC ~~train~~ loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2 2
4.4.1.4.1.1 DOC L02	SR 3.4.7.2	Verify required SG secondary side water level is ≥ ~~[25]~~ 10 %, narrow range	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 R1 2
DOC M02	SR 3.4.7.3	----------NOTE---------- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to each required SDC pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.1.4.1.3	SR 3.4.7.4	Verify required SDC ~~train~~ locations susceptible to gas accumulation are sufficiently filled with water. loop	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2 2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

3.4.1.4.1 LCO 3.4.7 One shutdown cooling (SDC) ~~train~~ loop shall be OPERABLE and in operation and either: 1

a. One additional SDC ~~train~~ loop shall be OPERABLE or 1

b. The secondary side water level of each steam generator (SG) shall be ≥ ~~[25~~ 10 narrow range %~~]~~. DOC L02 2 R1

---NOTES---

3.4.1.4.1 Note * 1. The SDC pump of the ~~train~~ loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided: 1

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1 , "SHUTDOWN MARGIN (SDM)," and 1

b. Core outlet temperature is maintained at ≥ 10°F below saturation temperature.

3.4.1.4.1 Note # 2. One SDC ~~train~~ loop may be inoperable for up to 2 hours for surveillance testing provided that the other SDC ~~train~~ loop is OPERABLE and in operation. 1

3.4.1.4.1 Note ## 3. No reactor coolant pump (RCP) shall be started with ~~any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR unless:~~ two idle RCS loops unless 1

3.4.14

~~a. The pressurizer water level is < [60]% or~~

~~b.~~ ~~The~~ secondary side water temperature in each SG is < ~~[100]~~ 40 °F above each of the RCS cold leg temperatures. 2

DOC L01 4. All SDC ~~trains~~ loops may not be in operation during planned heatup to MODE 4 when at least one RCS loop is in operation. 1

Applicability APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required SDC ~~train~~ loop inoperable. AND One SDC ~~train~~ loop OPERABLE.	A.1 Initiate action to restore a second SDC ~~train~~ loop to OPERABLE status. OR A.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately Immediately
B. One or more required SGs with secondary side water level not within limit. AND One SDC ~~train~~ loop OPERABLE.	B.1 Initiate action to restore a second SDC ~~train~~ loop to OPERABLE status. OR B.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately Immediately
C. No required SDC ~~trains~~ loops OPERABLE. OR Required SDC ~~train~~ loop not in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND C.2 Initiate action to restore one SDC ~~train~~ loop to OPERABLE status and operation.	Immediately Immediately

3.4.1.4.1 Action A, B
DOC A02
3.4.1.4.1 Action A, B
3.4.1.4.1 Action A, B
DOC A02
3.4.1.4.1 Action C
DOC M01
1
1
1
1
1
1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.1.4.1.2	SR 3.4.7.1	Verify required SDC ~~train~~ loop is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2 2
4.4.1.4.1.1 DOC L02	SR 3.4.7.2	Verify required SG secondary side water level is ≥ ~~[25]~~ 10 %, narrow range	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 R1 2
DOC M02	SR 3.4.7.3	----------NOTE---------- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to each required SDC pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.1.4.1.3	SR 3.4.7.4	Verify required SDC ~~train~~ locations susceptible to gas accumulation are sufficiently filled with water. loop	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.7, RCS LOOPS – MODE 5, LOOPS FILLED

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the shutdown cooling (SDC) heat exchangers. While the principal means for decay heat removal is via the SDC System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary side water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the SDC ~~trains~~ are the principal means for decay heat removal. The number of ~~trains~~ in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC ~~train~~ for decay heat removal and transport. The flow provided by one SDC ~~train~~ is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for decay heat removal. 1

The LCO provides for redundant paths of decay heat removal capability. The first path can be an SDC ~~train~~ that must be OPERABLE and in operation. The second path can be another OPERABLE SDC ~~train~~, or through the SGs via natural circulation (Ref. 1), each having an adequate water level. 1

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The SDC ~~trains~~ provide this circulation. 1

RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The purpose of this LCO is to require at least one of the SDC ~~trains~~ be OPERABLE and in operation with the other SDC ~~train~~ OPERABLE or secondary side water level of each SG shall be ≥ ~~[25]~~%. One SDC ~~train~~ provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second SDC ~~train~~ is normally maintained OPERABLE as a backup to the operating SDC ~~train~~ to provide redundant paths for decay heat removal. However, if the standby SDC ~~train~~ is not OPERABLE, a sufficient alternate method to provide redundant paths for decay heat removal is two SGs with their secondary side water levels ≥ ~~[25%]~~. Should the operating SDC ~~train~~ fail, the SGs could be used to remove the decay heat via natural circulation.

Note 1 permits all SDC pumps to be removed from operation ≤ 1 hour per 8 hour period. The circumstances for stopping both SDC ~~trains~~ are to be limited to situations where pressure and temperature increases can be maintained well within the allowable pressure (pressure and temperature and low temperature overpressure protection) and 10°F subcooling limits, or an alternate heat removal path through the SG(s) is in operation.

This LCO is modified by a Note that prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when SDC forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature, so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the SG(s) can be used as the backup for SDC heat removal. To ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

In MODE 5, it is sometimes necessary to stop all ~~RCP or~~ SDC forced circulation. This is permitted to change operation from one SDC ~~train~~ to the other, perform surveillance or startup testing, perform the transition to and from the SDC~~, or to avoid operation below the RCP minimum net positive suction head limit~~. The time period is acceptable because natural circulation is acceptable for decay heat removal, the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Note 2 allows one SDC ~~train~~ to be inoperable for a period of up to 2 hours provided that the other SDC ~~train~~ is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable ~~train~~ during the only time when such testing is safe and possible.

BASES

LCO (continued)

two idle RCS loops, the

Note 3 requires that ~~either of the following two conditions be satisfied~~ before an RCP may be started with ~~any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR:~~ 1

~~a. Pressurizer water level must be < [60]% or~~

~~b.~~ Secondary side water temperature in each SG must be < ~~[100]~~ 30 °F above each of the RCS cold leg temperatures. 2

Satisfying ~~either of~~ the ~~above~~ condition~~s~~ will preclude a low temperature overpressure event due to a thermal transient when the RCP is started. 1

loops

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting SDC ~~trains~~ to not be in operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the SDC ~~trains~~. 1

loops

loop

An OPERABLE SDC ~~train~~ is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. Management of gas voids is important to SDC System OPERABILITY. 1

SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

APPLICABILITY

loop

In MODE 5 with RCS loops filled, this LCO requires forced circulation to remove decay heat from the core and to provide proper boron mixing. One SDC ~~train~~ provides sufficient circulation for these purposes. 1

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2,"
LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.6, "RCS Loops – MODE 4,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and 3
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~. 3

BASES

ACTIONS

A.1, A.2, B.1 and B.2

If one SDC ~~train~~ is OPERABLE and any required SGs has secondary side water levels < ~~[25%]~~, or one required SDC ~~train~~ is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second SDC ~~train~~ to OPERABLE status or to restore the water level in the required SGs. Either Required Action will restore redundant decay heat removal paths. The immediate Completion Times reflect the importance of maintaining the availability of two paths for decay heat removal.

C.1 and C.2

If a required SDC ~~train~~ is not in operation, or no required SDC ~~train~~ is OPERABLE, except as permitted in Note 1, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended. Action to restore one SDC ~~train~~ to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification that one SDC ~~train~~ is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. ~~[The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

The SDC flow is established to ensure that core outlet temperature is maintained sufficiently below saturation to allow time for swapover to the pg 195standby SDC ~~train~~ should the operating ~~train~~ be lost.

loop

loop

SR 3.4.7.2

narrow range

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ ~~[25%]~~ ensures that redundant heat removal paths are available if the second SDC ~~train~~ is inoperable. The Surveillance is required to be performed when the LCO requirement is being met by use of the SGs. If both SDC ~~trains~~ are OPERABLE, this SR is not needed. ~~[The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.~~

10

loop

loops

R1

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.4.7.3

Verification that each required SDC ~~train~~ is OPERABLE ensures that redundant paths for decay heat removal are available. The requirement also ensures that the additional ~~train~~ can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

loop

loop

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The Surveillance is required to be performed when the LCO requirement is being met by one of two SDC trains, e.g., both SGs have < [25]% water level. [The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 1 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

SR 3.4.7.4

SDC System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the required SDC ~~train(s)~~ loop(s) and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel. 1

Selection of SDC System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The SDC System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the SDC System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

SDC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the SDC System piping and the procedural controls governing system operation.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the shutdown cooling (SDC) heat exchangers. While the principal means for decay heat removal is via the SDC System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary side water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the SDC ~~trains~~ are the principal means for decay heat removal. The number of ~~trains~~ in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC ~~train~~ for decay heat removal and transport. The flow provided by one SDC ~~train~~ is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for decay heat removal. 1

The LCO provides for redundant paths of decay heat removal capability. The first path can be an SDC ~~train~~ that must be OPERABLE and in operation. The second path can be another OPERABLE SDC ~~train~~, or through the SGs via natural circulation (Ref. 1), each having an adequate water level. 1

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The SDC ~~trains~~ provide this circulation. 1

RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The purpose of this LCO is to require at least one of the SDC ~~trains~~ be OPERABLE and in operation with the other SDC ~~train~~ OPERABLE or secondary side water level of each SG shall be ≥ ~~[25]~~%. One SDC ~~train~~ provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second SDC ~~train~~ is normally maintained OPERABLE as a backup to the operating SDC ~~train~~ to provide redundant paths for decay heat removal. However, if the standby SDC ~~train~~ is not OPERABLE, a sufficient alternate method to provide redundant paths for decay heat removal is two SGs with their secondary side water levels ≥ ~~[25%]~~. Should the operating SDC ~~train~~ fail, the SGs could be used to remove the decay heat via natural circulation.

Note 1 permits all SDC pumps to be removed from operation ≤ 1 hour per 8 hour period. The circumstances for stopping both SDC ~~trains~~ are to be limited to situations where pressure and temperature increases can be maintained well within the allowable pressure (pressure and temperature and low temperature overpressure protection) and 10°F subcooling limits, or an alternate heat removal path through the SG(s) is in operation.

This LCO is modified by a Note that prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when SDC forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained at least 10°F below saturation temperature, so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the SG(s) can be used as the backup for SDC heat removal. To ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

In MODE 5, it is sometimes necessary to stop all ~~RCP or~~ SDC forced circulation. This is permitted to change operation from one SDC ~~train~~ to the other, perform surveillance or startup testing, perform the transition to and from the SDC~~, or to avoid operation below the RCP minimum net positive suction head limit~~. The time period is acceptable because natural circulation is acceptable for decay heat removal, the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Note 2 allows one SDC ~~train~~ to be inoperable for a period of up to 2 hours provided that the other SDC ~~train~~ is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable ~~train~~ during the only time when such testing is safe and possible.

BASES

LCO (continued)

two idle RCS loops, the

Note 3 requires that ~~either of the following two conditions be satisfied~~ before an RCP may be started with ~~any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR:~~ 1

~~a. Pressurizer water level must be < [60]% or~~

~~b.~~ Secondary side water temperature in each SG must be < ~~[100]~~ 40 °F above each of the RCS cold leg temperatures. 2

Satisfying ~~either of~~ the ~~above~~ condition~~s~~ will preclude a low temperature overpressure event due to a thermal transient when the RCP is started. 1

loops

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting SDC ~~trains~~ to not be in operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the SDC ~~trains~~. 1

loops

loop

An OPERABLE SDC ~~train~~ is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. Management of gas voids is important to SDC System OPERABILITY. 1

SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

APPLICABILITY

loop

In MODE 5 with RCS loops filled, this LCO requires forced circulation to remove decay heat from the core and to provide proper boron mixing. One SDC ~~train~~ provides sufficient circulation for these purposes. 1

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2,"
LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.6, "RCS Loops – MODE 4,"
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and 3
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~. 3

BASES

ACTIONS

A.1, A.2, B.1 and B.2

If one SDC ~~train~~ is OPERABLE and any required SGs has secondary side water levels < ~~[25%]~~, or one required SDC ~~train~~ is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second SDC ~~train~~ to OPERABLE status or to restore the water level in the required SGs. Either Required Action will restore redundant decay heat removal paths. The immediate Completion Times reflect the importance of maintaining the availability of two paths for decay heat removal.

C.1 and C.2

If a required SDC ~~train~~ is not in operation, or no required SDC ~~train~~ is OPERABLE, except as permitted in Note 1, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended. Action to restore one SDC ~~train~~ to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification that one SDC ~~train~~ is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. ~~[The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

The SDC flow is established to ensure that core outlet temperature is maintained sufficiently below saturation to allow time for swapover to the standby SDC ~~train~~ should the operating ~~train~~ be lost.

loop

loop

SR 3.4.7.2

narrow range

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ ~~[25%]~~ ensures that redundant heat removal paths are available if the second SDC ~~train~~ is inoperable. The Surveillance is required to be performed when the LCO requirement is being met by use of the SGs. If both SDC ~~trains~~ are OPERABLE, this SR is not needed. | R1

10

loop

loops

~~[The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.4.7.3

Verification that each required SDC ~~train~~ is OPERABLE ensures that redundant paths for decay heat removal are available. The requirement also ensures that the additional ~~train~~ can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

loop

loop

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The Surveillance is required to be performed when the LCO requirement is being met by one of two SDC trains, e.g., both SGs have < [25]% water level. [The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 1 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

SR 3.4.7.4

SDC System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the required SDC ~~train(s)~~ loop(s) and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel. 1

Selection of SDC System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The SDC System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the SDC System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

SDC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the SDC System piping and the procedural controls governing system operation.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~---------- REVIEWER'S NOTE ----------~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~----------]~~

2

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.7, BASES, RCS LOOPS – MODE 5, LOOPS FILLED

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the correct Specification Title.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.7, RCS LOOPS – MODE 5, LOOPS FILLED

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 8

3.4.8, RCS Loops – MODE 5, Loops Not Filled

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTOR COOLANT SYSTEM

COLD SHUTDOWN – LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

LCO 3.4.8 ~~3.4.1.4.2~~ Two shutdown cooling (SDC) loops shall be OPERABLE# and ~~at least~~ one ~~shutdown cooling~~ SDC loop shall be in operation*.

Applicability **APPLICABILITY:** MODE 5 with ~~reactor coolant~~ RCS loops not filled.

ACTION:

ACTION A ~~a.~~ ~~With less than the above~~ One required SDC loop~~s~~ IN OPERABLE, ~~within one (1) hour~~ immediately M01 initiate ~~corrective~~ action to ~~return the required~~ restore loop~~s~~ not to OPERABLE status.

ACTION B No required SDC loop OPERABLE OR Required SDC ~~b.~~ ~~With no shutdown cooling~~ SDC loop in operation, immediately M01 suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of ~~Technical Specification~~ LCO 3.1.1~~.2~~ and ~~within one (1) hour~~ immediately M01 initiate ~~corrective~~ action to ~~return the required shutdown cooling~~ restore one SDC loop to operation. OPERABLE status and M01

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1 ~~4.4.1.4.2~~ ~~At least one shutdown cooling~~ Verify required SDC loop ~~shall be determined to be~~ is in operation ~~and circulating reactor coolant~~ in accordance with the Surveillance Frequency Control Program. LA01

SR 3.4.8.3 ~~4.4.1.4.2.1~~ Verify ~~shutdown cooling train~~ Insert proposed SR 3.4.8.2 M02 locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program. SDC loop

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LCO 3.4.8 Note 2 # One ~~shutdown cooling~~ SDC loop may be inoperable for ~~up to~~ ≤ 2 hours for surveillance testing provided the other ~~shutdown cooling~~ SDC loop is OPERABLE and in operation.

LCO 3.4.8 Note 1 * ~~The shutdown cooling~~ SDC pump may be ~~de-energized~~ removed from operation for up to 1 hour provided 1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of ~~Technical Specification~~ LCO 3.1.1~~.2~~ and 2) core outlet temperature is maintained ~~at least~~ > 10°F below saturation temperature. A02 | R1

Insert proposed LCO 3.4.8 Note 1.c M03 | R1

~~DELETED~~

REACTOR COOLANT SYSTEM

COLD SHUTDOWN – LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

LCO 3.4.8 ~~3.4.1.4.2~~ Two ~~shutdown cooling~~ (SDC) loops shall be OPERABLE# and ~~at least~~ one ~~shutdown cooling~~ SDC loop shall be in operation.*

Applicability **APPLICABILITY:** MODE 5 with ~~reactor coolant~~ RCS loops not filled.

ACTION:

ACTION A ~~a.~~ ~~With less than the above~~ One SDC IN required loop~~s~~ OPERABLE, ~~within 1 hour~~ immediately (M01) initiate ~~corrective~~ action to ~~return the required~~ restore loop~~s~~ to OPERABLE status ~~as soon as possible~~.

ACTION B ~~b.~~ ~~With no shutdown cooling~~ No required SDC loop OPERABLE OR Required SDC not in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of ~~Technical Specification~~ 3.1.1~~.2~~ and ~~within 1 hour~~ initiate ~~corrective~~ action to ~~return the required shutdown cooling~~ restore one SDC loop to operation. immediately (M01)

OPERABLE status and (M01)

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1 ~~4.4.1.4.2~~ ~~At least one shutdown cooling~~ Verify required SDC loop ~~shall be determined to be~~ is in operation ~~and circulating reactor coolant~~ in accordance with the Surveillance Frequency Control Program. (LA01)

Insert proposed SR 3.4.8.2 (M02)

SR 3.4.8.3 ~~4.4.1.4.2.1~~ Verify ~~shutdown cooling trains~~ locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

SDC loop

LCO 3.4.8 Note 2 # One ~~shutdown cooling~~ SDC loop may be inoperable for ~~up to~~ ≤ 2 hours for surveillance testing provided the other ~~shutdown cooling~~ loop is OPERABLE and in operation.

LCO 3.4.8 Note 1 * ~~The shutdown cooling~~ SDC pump~~s~~ may be ~~deenergized~~ removed from operation for up to 1 hour provided (1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of ~~Technical Specification~~ LCO 3.1.1~~.2~~ and (2) core outlet temperature is maintained ~~at least~~ > 10°F below saturation temperature. (A02) | R1

Insert proposed LCO 3.4.8 Note 1.c (M03) | R1

~~DELETED~~

DISCUSSION OF CHANGES
ITS 3.4.8, RCS LOOPS – MODE 5, LOOPS NOT FILLED

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.1.4.2 states, in part, that two shutdown cooling (SDC) loops shall be OPERABLE and at least one SDC loop shall be in operation. CTS 3.4.1.4.2 is modified by footnote * that states, in part, that the SDC pump "may be de-energized" for up to 1 hour. ITS 3.4.8 NOTE 1 states that the pumps "may be removed from operation." This changes the CTS by changing "de-energized" to "removed from operation." R1

The purpose of the NOTE is to allow for stopping flow for up to the specified period of 1 hour. This change is a clarification of the intended action to stop the pumps rather than fully de-energize the pumps (e.g., for a maintenance activity). This change is acceptable because it is consistent with the application of CTS footnote *. This change is designated as administrative because it does not result in a technical change.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.1.4.2 Action a. states, in part, with less than the above required loops OPERABLE, within one (1) hour initiate corrective action to return the required loops to OPERABLE status. ITS 3.4.8 ACTION A states that with one required SDC loop inoperable "immediately" initiate action to restore SDC loop to OPERABLE status. This changes the CTS by adding the requirement to "immediately" initiate corrective action to return the required loops to OPERABLE status.

CTS 3.4.1.4.2 Action b. states, with no shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specification 3.1.1.2 and within one (1) hour initiate corrective action to return the required shutdown cooling loop to operation. ITS 3.4.8 ACTION B states that with no required SDC loop OPERABLE or required SDC loop not in operation, "immediately" suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1, and "immediately" initiate action to restore one SDC loop to OPERABLE status and operation. This changes the CTS by adding the requirement to "immediately" suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1 and to "immediately" initiate corrective action to return the

required shutdown cooling loop to operation. Additionally, this changes the CTS by adding the requirement to restore the loop to OPERABLE status.

This change is acceptable because it provides appropriate actions if one required SDC loop is inoperable, no required SDC loop is OPERABLE, or a required SDC loop is not in operation. Under these conditions immediate action is necessary to ensure unit transients do not occur, and action is immediately taken to restore SDC to OPERABLE status and operation to be able to remove the decay heat generated by the reactor. These actions are required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operation. Action to restore one SDC loop to OPERABLE status and operation is necessary to be able to remove the decay heat generated by the reactor. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. This change is designated as more restrictive, because it requires immediate action to return the required reactor coolant loop to operation and adds the requirement to restore the loop to OPERABLE status.

M02 ITS SR 3.4.8.2 requires verification that correct breaker alignment and indicated power are available to each required SDC pump. A Note further explains that the Surveillance is not required to be performed until 24 hours after a required pump is not in operation. This Surveillance is not required by the CTS. This changes the CTS by requiring verification of correct breaker alignment and indicated power availability on required SDC pumps that are not in operation.

The purpose of ITS SR 3.4.8.2 is to ensure a standby SDC pump is available to provide RCS cooling should the operating pump fail. This change is acceptable because the verification of proper breaker alignment and power availability ensures that an additional SDC pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. This change is designated as more restrictive because it adds a Surveillance Requirement to the CTS.

M03 CTS 3.4.1.4.2 states, in part, that two shutdown cooling loops shall be OPERABLE and at least one shutdown cooling loop shall be in operation. CTS 3.4.1.4.2 is modified by a footnote * that states, in part, that the shutdown cooling pump may be de-energized for up to 1 hour under specified conditions. ITS 3.4.8 Note 1 allows all SDC pumps to be removed from operation “when switching from one loop to another” under specified conditions (i.e., part 1.a and 1.b which are the same as the conditions specified in the CTS) and also requires that no draining operations to further reduce the RCS water volume are permitted (part c). This changes the CTS by adding a restriction that “no draining operations to further reduce the RCS water volume are permitted.” R1 R1

The purpose of the CTS 3.4.1.4.2 Footnote * in MODE 5 with RCS loops not filled is to allow the SDC loops to be removed from operation to allow sufficient time to perform activities required to refill the RCS loops. Specifically, the SDC pumps are removed from service to allow reactor coolant pump operation to remove voids from the steam generator U-tubes. This change is acceptable because ITS LCO 3.4.8 Note 1 provides sufficient time to perform RCS loop refill R1

operations and provides adequate controls. Stopping all operating SDC loops when the RCS is not filled is limited to short periods of time because of the reduced inventory of water available to absorb decay heat. The additional condition that no draining operations be performed during the switching evolution is reasonable given the potential for low RCS water level. This change is more restrictive because it adds an additional restriction. R1

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.1.4.2 states, in part, at least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant. ITS SR 3.4.8.1 states verify required SDC loop is in operation. This changes the CTS by moving the Surveillance Requirement to verify that the SDC loops are circulating reactor coolant, to the Bases.

The removal of this detail for determining LCO requirements and performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be in the Technical Specifications in order to provide adequate protection of the public health and safety. The ITS retains the requirement that two shutdown cooling (SDC) loops shall be OPERABLE and one SDC loop shall be in operation. Any loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

3.4.1.4.2 LCO 3.4.8 Two shutdown cooling (SDC) ~~trains~~ loops shall be OPERABLE and one SDC ~~train~~ loop shall be in operation. 1

----------NOTES----------

3.4.1.4.2 Note * 1. All SDC pumps may be removed from operation for ≤ ~~15 minutes when switching from one train to another~~ 1 hour provided: 3 | R1

~~[~~ a. The core outlet temperature is maintained > 10°F below saturation temperature, ~~]~~ 2

b. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1 and

DOC M03 c. No draining operations to further reduce the RCS water volume are permitted.

3.4.1.4.2 Note # 2. One SDC ~~train~~ loop may be inoperable for ≤ 2 hours for surveillance testing provided the other SDC ~~train~~ loop is OPERABLE and in operation. 1

Applicability APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.1.4.2.a DOC M01 A. One required SDC ~~train~~ loop inoperable.	A.1 Initiate action to restore SDC ~~train~~ loop to OPERABLE status.	Immediately

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No required SDC ~~train~~ loop OPERABLE. OR Required SDC ~~train~~ loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND B.2 Initiate action loop to restore one SDC ~~train~~ to OPERABLE status and operation.	Immediately Immediately

3.4.1.4.2.b DOC M01

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify required SDC loop ~~train~~ is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.4.1.4.2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
DOC M02	SR 3.4.8.2	----------NOTE---------- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to each required SDC pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.1.4.2.1	SR 3.4.8.3	Verify SDC ~~train~~ loop locations susceptible to gas accumulation are sufficiently filled with water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2 2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

3.4.1.4.2 LCO 3.4.8 Two shutdown cooling (SDC) ~~trains~~ loops shall be OPERABLE and one SDC ~~train~~ loop shall be in operation. (1)

----------NOTES----------

3.4.1.4.2 Note *

1. All SDC pumps may be removed from operation for ≤ ~~15 minutes when switching from one train to another~~ 1 hour provided: (3) | R1

   ~~[~~ a. The core outlet temperature is maintained > 10°F below saturation temperature, ~~]~~ (2)

   b. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1 and

DOC M03

   c. No draining operations to further reduce the RCS water volume are permitted.

3.4.1.4.2 Note #

2. One SDC ~~train~~ loop may be inoperable for ≤ 2 hours for surveillance testing provided the other SDC ~~train~~ loop is OPERABLE and in operation. (1)

Applicability APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.1.4.2.a DOC M01 A. One required SDC ~~train~~ loop inoperable.	A.1 Initiate action to restore SDC ~~train~~ loop to OPERABLE status.	Immediately (1)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No required SDC ~~train~~ loop OPERABLE. OR Required SDC ~~train~~ loop not in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1. AND B.2 Initiate action loop to restore one SDC ~~train~~ to OPERABLE status and operation.	Immediately Immediately

3.4.1.4.2.b DOC M01

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify required SDC loop ~~train~~ is in operation.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.4.1.4.2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
DOC M02	SR 3.4.8.2	----------NOTE---------- Not required to be performed until 24 hours after a required pump is not in operation. ---------- Verify correct breaker alignment and indicated power available to each required SDC pump.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.1.4.2.1	SR 3.4.8.3	Verify SDC ~~train~~ loop locations susceptible to gas accumulation are sufficiently filled with water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.8, RCS LOOPS – MODE 5, LOOPS NOT FILLED

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ISTS LCO 3.4.8 NOTE 1 allows all shutdown cooling (SDC) pumps to be "removed from operation for ≤ 15 minutes when switching from one train to another." ISTS LCO 3.4.8 is modified consistent with CTS 3.4.1.4.2 allowing the SDC pumps to be removed from operation for ≤ 1 hour. R1

   This change to the ISTS is consistent with the CTS, which allows the SDC pumps to be "de-energized" for up to 1 hour under specific conditions. Maintaining the CTS allowance is necessary to allow sufficient time to perform activities required to refill the RCS loops. Specifically, the SDC pumps are removed from service to allow reactor coolant pump operation to remove voids from the steam generator (SG) U-tubes. Refilling the RCS loops and removing voids from the SG U-tubes provide the added benefit of thermal coupling between the RCS and SGs, which will provide an additional decay heat removal method (i.e., SG cooling). This deviation from the ISTS has also been previously approved for several plants. For example, Arkansas Nuclear One, Unit 1 and Wolf Creek Unit 1 allow the decay heat removal/residual heat removal pumps to be removed from operation for ≤ 1 hour (Refer to ADAMS Accession No. ML053130314 and MO 052720315, respectively), and Palo Verde, Units 1, 2, and 3 allow the SDC pumps to be removed from operation ≤ 1 hour per 8 hour period (Refer to ADAMS Accession No. ML053110045).

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the shutdown cooling (SDC) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only the SDC System can be used for coolant circulation. The number of ~~trains~~ loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC ~~train~~ loop for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal. 1 1

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in determining the time available for mitigation of the accidental boron dilution event. The SDC ~~trains~~ loops provide this circulation. The flow provided by one SDC ~~train~~ loop is adequate for decay heat removal and for boron mixing. 1

RCS loops - MODE 5 (loops not filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require a minimum of two SDC ~~trains~~ loops be OPERABLE and one of these ~~trains~~ loops be in operation. An OPERABLE ~~train~~ loop is one that is capable of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the SDC System unless forced flow is used. A minimum of one running SDC pump meets the LCO requirement for one ~~train~~ loop in operation. An additional SDC ~~train~~ loop is required to be OPERABLE to meet the single failure criterion. 1 1

Note 1 permits the SDC pumps to be removed from operation for ≤ ~~15 minutes when switching from one train to another~~ 1 hour. The circumstances for stopping both SDC pumps are to be limited to situations when the outage time is short ~~[~~and the core outlet temperature is maintained > 10°F below saturation temperature~~]~~. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained or draining operations when SDC forced flow is stopped. 4 R1 2

Note 2 allows one SDC ~~train~~ loop to be inoperable for a period of 2 hours provided that the other ~~train~~ loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable ~~train~~ loop during the only time when these tests are safe and possible. 1

BASES

LCO (continued)

loop

An OPERABLE SDC ~~train~~ is composed of an OPERABLE SDC pump capable of providing forced flow to an OPERABLE SDC heat exchanger, along with the appropriate flow and temperature instrumentation for control, protection, and indication. SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. Management of gas voids is important to SDC System OPERABILITY. 1

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the SDC System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2,"
LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.6, "RCS Loops - MODE 4,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and 3
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~. 3

ACTIONS

A.1

loop

If one required SDC ~~train~~ is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second ~~train~~ to OPERABLE status. The Completion Time reflects the importance of maintaining the availability of two paths for heat removal. 1

loop

B.1 and B.2

loop

If no required SDC ~~train~~ is OPERABLE or the required ~~train~~ is not in operation, except as provided in Note 1, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended. 1

loop

Action to restore one SDC ~~train~~ to OPERABLE status and operation must be initiated immediately. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification that the required SDC ~~train~~ is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. ~~[The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions.~~

loop

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

2

SR 3.4.8.2

loop

loop

Verification that each required ~~train~~ is OPERABLE ensures that redundant paths for heat removal are available and that an additional ~~train~~ can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and indicated power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

2

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

SR 3.4.8.3

SDC System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the SDC ~~trains~~ loops and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel. 1

Selection of SDC System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The SDC System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the SDC System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

SDC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating

BASES

SURVEILLANCE REQUIREMENTS (continued)

parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the SDC System piping and the procedural controls governing system operation.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

REFERENCES None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the shutdown cooling (SDC) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only the SDC System can be used for coolant circulation. The number of ~~trains~~ loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one SDC ~~train~~ loop for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal. 1 1

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in determining the time available for mitigation of the accidental boron dilution event. The SDC ~~trains~~ loops provide this circulation. The flow provided by one SDC ~~train~~ loop is adequate for decay heat removal and for boron mixing. 1

RCS loops - MODE 5 (loops not filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require a minimum of two SDC ~~trains~~ loops be OPERABLE and one of these ~~trains~~ loops be in operation. An OPERABLE ~~train~~ loop is one that is capable of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the SDC System unless forced flow is used. A minimum of one running SDC pump meets the LCO requirement for one ~~train~~ loop in operation. An additional SDC ~~train~~ loop is required to be OPERABLE to meet the single failure criterion. 1 1

Note 1 permits the SDC pumps to be removed from operation for ≤ 1 hour ~~15 minutes when switching from one train to another~~. The circumstances for stopping both SDC pumps are to be limited to situations when the outage time is short ~~[~~and the core outlet temperature is maintained > 10°F below saturation temperature~~]~~. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained or draining operations when SDC forced flow is stopped. 4 | R1 2

Note 2 allows one SDC ~~train~~ loop to be inoperable for a period of 2 hours provided that the other ~~train~~ loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable ~~train~~ loop during the only time when these tests are safe and possible. 1

BASES

LCO (continued)

loop

An OPERABLE SDC ~~train~~ is composed of an OPERABLE SDC pump capable of providing forced flow to an OPERABLE SDC heat exchanger, along with the appropriate flow and temperature instrumentation for control, protection, and indication. SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. Management of gas voids is important to SDC System OPERABILITY.

1

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the SDC System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2,"
LCO 3.4.5, "RCS Loops - MODE 3,"
LCO 3.4.6, "RCS Loops - MODE 4,"
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level" ~~(MODE 6)~~, and
LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level" ~~(MODE 6)~~.

3

3

ACTIONS

A.1

loop

If one required SDC ~~train~~ is inoperable, redundancy for heat removal is lost. Action must be initiated immediately to restore a second ~~train~~ to OPERABLE status. The Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

loop

1

B.1 and B.2

loop

If no required SDC ~~train~~ is OPERABLE or the required ~~train~~ is not in operation, except as provided in Note 1, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended.

loop

Action to restore one SDC ~~train~~ to OPERABLE status and operation must be initiated immediately. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

loop

1

1

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification that the required SDC ~~train~~ is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. ~~[The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions.~~

loop

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

2

SR 3.4.8.2

loop

loop

Verification that each required ~~train~~ is OPERABLE ensures that redundant paths for heat removal are available and that an additional ~~train~~ can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and indicated power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

2

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

SR 3.4.8.3

SDC System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the SDC ~~trains~~ loops and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel. 1

Selection of SDC System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The SDC System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the SDC System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

SDC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating

BASES

SURVEILLANCE REQUIREMENTS (continued)

parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the SDC System piping and the procedural controls governing system operation.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

REFERENCES None.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.8, BASES, RCS LOOPS – MODE 5, LOOPS NOT FILLED

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the Specification Title.

4. This ISTS Bases change reflects the change to the Specification, which permits the SDC pumps to be removed from operation for ≤ 1 hour. This change represents a retention of the CTS criteria. This deviation from the ISTS has also been previously approved for several plants. For example, Arkansas Nuclear One, Unit 1 and Wolf Creek Unit 1 allow the decay heat removal/residual heat removal pumps to be removed from operation for ≤ 1 hour (Refer to ADAMS Accession No. ML053130314 and MO 052720315, respectively), and Palo Verde, Units 1, 2, and 3 allow the SDC pumps to be removed from operation ≤ 1 hour per 8 hour period (Refer to ADAMS Accession No. ML053110045). R1

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.8, RCS LOOPS – MODE 5, LOOPS NOT FILLED

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 9

3.4.9, Pressurizer

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

pressurizer heaters OPERABLE with the capacity of ≥ — Pressurizer water level ≤ 68% — A02

LCO 3.4.9

~~3.4.4~~ The pressurizer shall be OPERABLE with ~~a steam bubble~~, and ~~with at least~~ 150 kw ~~of pressurizer heaters~~ capable of ~~being supplied by~~ emergency power.

and — being powered from an — supply

Applicability

APPLICABILITY: MODES 1 and 2. — Insert proposed MODE 3 Applicability — M01

ACTION:

water level not within limit — MODE 3

ACTION A

~~With the~~ pressurizer ~~inoperable~~, be in ~~at least HOT STANDBY~~ with ~~the~~ reactor trip breakers open within 6 hours.

Insert proposed Required Action A.2 — M01

Insert proposed ACTIONS B and C — L02 — M01

SURVEILLANCE REQUIREMENTS

~~4.4.4~~ ~~In accordance with 4.8.1.1.2 .~~ — A03

Insert proposed SR 3.4.9.1 — M02

Insert proposed SR 3.4.9.2 — M02

Insert proposed SR 3.4.9.3 — M03

A01

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

LCO 3.4.9 ~~3.4.3~~ The pressurizer shall be OPERABLE with a ~~minimum~~ water level ~~of greater than or equal to 27% indicated level and a maximum water level of less than or equal to~~ 68% ~~indicated level~~ and ~~at least~~ two groups of pressurizer heaters capable of being powered from ~~1E buses each having a nominal~~ capacity of ~~at least~~ 150 kW.

LCO 3.4.9.a

LCO 3.4.9.b

Pressurizer

≤

an emergency power supply

with the

each group ≥

L01

Applicability **APPLICABILITY:** MODES 1, 2 and 3.

ACTION:

ACTION B ~~a.~~ ~~With~~ one group of ~~the above~~ required pressurizer heaters inoperable, restore ~~at least two~~ groups to OPERABLE status within 72 hours or be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours.

required

of pressurizer heaters

ACTION D

MODE 3

MODE 4

12

Condition C Note

NOTE

Action not applicable when second group of required pressurizer heaters intentionally made inoperable.

ACTION C ~~b.~~ ~~With~~ two groups of required pressurizer heaters inoperable, restore at least one group of required pressurizer heaters to OPERABLE status within 24 hours or be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours.

ACTION D

MODE 3

MODE 4

12

water level not within limit

ACTION A ~~c.~~ ~~With the~~ pressurizer ~~otherwise inoperable~~, be in ~~at least HOT STANDBY~~ with ~~the~~ reactor trip breakers open within 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours.

MODE 3

MODE 4

12

A04

SURVEILLANCE REQUIREMENTS

Verify

level is ≤ 68%

SR 3.4.9.1 ~~4.4.3.1~~ ~~The~~ pressurizer ~~water volume shall be determined to be within its limits~~ in accordance with the Surveillance Frequency Control Program.

Verify

SR 3.4.9.2 ~~4.4.3.2~~ ~~The~~ capacity of each ~~of the above~~ required groups of pressurizer heaters ~~shall be verified to be at least~~ 150 kW in accordance with the Surveillance Frequency Control Program.

≥

are capable of being powered from an

Verify required

~~4.4.3.3~~ ~~The~~ emergency power supply ~~for the~~ pressurizer heaters ~~shall be demonstrated OPERABLE~~ in accordance with the Surveillance Frequency Control Program ~~by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power:~~

A03

~~a. the pressurizer heaters are automatically shed from the emergency power sources, and~~

~~b. the pressurizer heaters can be reconnected to their respective buses manually from the control room after resetting of the ESFAS test signal.~~

LA01

DISCUSSION OF CHANGES
ITS 3.4.9, PRESSURIZER

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 1 only:** CTS 3.4.4 states that the pressurizer shall be OPERABLE with a steam bubble. ITS 3.4.9 requires that the pressurizer water level be ≤ 68% to be OPERABLE. This changes the CTS by replacing the generic statement, "…with a steam bubble," with a specific pressurizer water level requirement.

This change is acceptable because when the unit is in MODE 1, 2, or 3, maintaining pressurizer water level ≤ 68% ensures a steam bubble exists. Since the ITS continues to require a steam bubble (i.e., pressurizer water level ≤ 68%) there is no need to require a steam bubble in the pressurizer. The change is designated as administrative because it does not result in a technical change to the CTS but clarifies the maximum level to support a steam bubble.

A03 Unit 1 CTS 4.4.4 states the Surveillance is in accordance with 4.8.1.1.2. Unit 2 CTS 4.4.3.3.a states, in part, that the emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE by verifying that upon an safety injection actuation signal (SIAS) test signal concurrent with a loss of offsite power, the pressurizer heaters are automatically shed from the emergency power sources. Unit 1 CTS 4.8.1.1.2.e.5 and Unit 2 CTS 4.8.1.1.2.e.6 verify that upon an actual or simulated loss of offsite power in conjunction with an SIAS actuation signal, load shedding from the emergency busses occurs. Similarly, ITS SR 3.8.1.17 verifies that upon an actual or simulated loss of offsite power in conjunction with an SIAS actuation signal, load shedding from the emergency busses occurs.

The purpose of Unit 1 CTS 4.4.4 and Unit 2 CTS 4.4.3.3.a is to demonstrate that the pressurizer heaters are automatically shed from the emergency power supplies upon an actual or simulated loss of offsite power in conjunction with an SIAS actuation signal. This requirement is addressed in Unit 1 CTS 4.8.1.1.2.e.5 and Unit 2 CTS 4.8.1.1.2.e.6 Surveillances, and in ITS Unit 1 and Unit 2 SR 3.8.1.17 Surveillances. This change is acceptable because the condition under which the CTS Surveillance applies has not changed. This change is designated as administrative as it results in no technical change to the CTS.

A04 **Unit 2 Only:** CTS 3.4.3 states the LCO requirements for pressurizer water level and pressurizer heater capacity. CTS 3.4.3 Actions a. and b. provide actions for one group of required pressurizer heaters inoperable and two groups of required pressurizer heaters, respectively. ITS 3.4.9 similarly, provides ACTIONS B, C, and D for inoperable group(s) of required pressure heaters. CTS 3.4.3 Action c. states, in part, with the "pressurizer otherwise inoperable" be in at least HOT

STANDBY (i.e., MODE 3) with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN (i.e., MODE 4) within the following 6 hours. Similarly, ITS 3.4.9 ACTION A states that with the pressurizer level not within limit, be in MODE 3 with the reactor trip breakers open within 6 hours and MODE 4 within 12 hours. This changes the CTS to specifically state the reason the pressurizer is inoperable for reasons other than pressurizer heaters.

The purpose of CTS 3.4.3 is to require the pressurizer to be OPERABLE and two conditions of OPERABILITY are supplied. The conditions are pressurizer water level and pressurizer heater OPERABILITY. CTS 3.4.4 Action c. only applies when water level is not within limit. This is the same condition for which ITS 3.4.9 Condition A applies. This change is acceptable because the condition under which CTS 3.4.3 Action c. applies has not changed. This change is designated as administrative as it results in no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS 3.4.4 only requires the pressurizer to be OPERABLE in MODES 1 and 2. If the pressurizer is inoperable, the CTS Action requires the unit be in at least HOT STANDBY (i.e., MODE 3) with the reactor trip breakers open within 6 hours. ITS 3.4.9 requires the pressurizer to be OPERABLE in MODES 1, 2, and 3. ITS ACTION A requires that if pressurizer water level is not within limits, the unit must be in MODE 3 with reactor trip breakers open within 6 hours and in MODE 4 within 12 hours. ITS ACTION C requires that if the heater capacity is not within limits and is not restored to OPERABLE status within 24 hours, the unit must be in MODE 3 within 6 hours and in MODE 4 within 12 hours. ITS 3.4.9 Required Action A.2 and Required Action C.2 each require the unit be in MODE 4 within 12 hours. This changes the CTS by expanding the Applicability of the Pressurizer to include MODE 3 and requiring the unit to exit this new Applicability within 12 hours when pressurizer water level is not within limits, or when pressurizer heater capacity is not within limits and is not restored to within limits within 24 hours.

The purpose of the ITS MODE 3 Applicability is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation. This change is acceptable because it provides appropriate requirements in MODE 3 to achieve this purpose. This change is designated as more restrictive because it requires the pressurizer and associated heaters to be OPERABLE under more conditions than is currently required.

M02 **Unit 1 only:** CTS 4.4.4 requires testing in accordance with CTS 4.8.1.1.2. ITS proposes additional Surveillances to verify the LCO is met. ITS SR 3.4.9.1 requires verification that the pressurizer water level is ≤ 68% and ITS SR 3.4.9.2 requires verification that the required pressurizer heater capacity is ≥ 150 kW. This changes the CTS by adding additional Surveillances.

This change is necessary to periodically verify pressurizer level is within the required limit to ensure conformance with a safety analysis assumption and to periodically verify the required pressurizer heater group has adequate capability to perform its function. The Frequencies for these SRs will be in accordance with

the Surveillance Frequency Control Program. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per CTS 6.8.4.o. Therefore, SR 3.4.9.1, and SR 3.4.9.2 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial 12 hour Frequency for SR 3.4.9.1, and an initial 18 month Frequency for SR 3.4.9.2, consistent with the ISTS SR 3.4.9.1 and SR 3.4.9.2, respectively.

M03 **Unit 1 only:** CTS 4.4.4 states that Surveillance Requirements are in accordance with 4.8.1.1.2. CTS 4.8.2.1.1 demonstrates the emergency power supplies for the pressurizer heaters are OPERABLE by verifying that on an SIAS test signal concurrent with a loss of offsite power, the pressurizer heaters shed from the emergency power supplies. ITS SR 3.4.9.3 demonstrates that the heaters are capable of being powered from an emergency power supply. This changes the CTS by adding a Surveillance. The Frequency for this SR will be in accordance with the Surveillance Frequency Control Program. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per CTS 6.8.4.o. Therefore, SR 3.4.9.3 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 18 months consistent with the ISTS SR 3.4.9.3.

The purpose of the Surveillance, in part, is to demonstrate that the pressurizer heaters can be manually transferred to and energized by emergency power supplies after reset of the SIAS. ITS SR 3.4.9.3 is added to establish the requirement that the pressurizer heaters be capable of being powered from an emergency power supply, which will also demonstrate that the required pressurizer heater group can be reconnected to its respective bus manually from the control room after resetting the SIAS test signal. This change is designated as a more restrictive because a Surveillance is added to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 **Unit 2 only:** *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.3.3 states, in part, that the emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE by verifying that on an SIAS test signal concurrent with a loss of offsite power, the pressurizer heaters which have been shed from the emergency power supplies, can be reconnected to their respective buses manually from the control room after resetting of the SIAS test signal. ITS SR 3.4.9.3 requires that pressurizer heaters are capable of being powered from an emergency power supply. ITS SR 3.4.9.3 does not contain the detail for pressurizer heaters reconnection to emergency power. This changes the CTS by deleting these procedural details.

The purpose of CTS 4.4.3.3. is to demonstrate that the pressurizer heaters can be manually transferred to and energized by emergency power supplies after

reset of the SIAS. The requirement that the pressurizer heaters be automatically shed from the emergency power sources upon receipt of an SIAS test signal is provided in LCO 3.8.1 Surveillances. ITS SR 3.4.9.3 retains the requirement that the pressurizer heaters are capable of being powered from an emergency power supply. The requirement that the pressurizer heaters can be reconnected to their respective emergency buses manually after reset of the SIAS test signal is relocated to the ITS Bases, which states the Surveillance demonstrates that the heaters can be manually transferred to and energized by emergency power supplies following an actual or simulated loss of offsite power concurrent with an SIAS actuation signal. This change is acceptable because this type of procedural detail will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 **Unit 2 only:** *(Category 1 - Relaxation of LCO Requirements)* CTS 3.4.3 states, in part, that the pressurizer shall be OPERABLE with a minimum water level of ≥ 27% indicated level and a maximum water level of ≤ 68% indicated level. ITS LCO 3.4.9.a states that the pressurizer shall be OPERABLE with pressurizer water level ≤ 68%. This changes the CTS by eliminating the lower water level limit of ≥ 27%.

The purpose of the CTS 3.4.3 lower limit is to preserve the steam space during normal operation, allowing both sprays and heaters to maintain the design operating pressure. The lower level limit prevents the low level interlock from deenergizing the pressurizer heaters during steady state operations. This change is acceptable because the low water level limit is not necessary for accident mitigation and is not a process variable that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The pressurizer water level is routinely monitored by operations personnel to ensure a low level in the pressurizer does not occur, similar to other plant parameters not specified in the Technical Specifications. Therefore, the low level limit is not necessary to be included in the Technical Specifications. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than are being applied in the CTS.

L02 **Unit 1 only:** *(Category 3 - Relaxation of Completion Time)* CTS 3.4.4 does not contain separate Actions for the pressurizer water level and the pressurizer heaters. ITS provides separate ACTIONS (A and B) for LCO 3.4.9.a (pressurizer water level) and LCO 3.4.9.b (pressurizer heaters capacity), respectively. ITS 3.4.9 ACTIONS A and B provide compensatory measures when the pressurizer water level > 68% (ACTION A) and when the capacity of the required pressurizer heaters is less than 150 kW (ACTION B). ITS 3.49 ACTION A requires the unit be in MODE 3 with reactor trip breakers open within 6 hours, and in MODE 4 within 12 hours. ITS 3.4.9 ACTION B requires restoration of the pressurizer

heater capability within 24 hours. If the required pressurizer heaters are not restored within 24 hours, ITS 3.4.9 ACTION C requires the unit be in MODE 3 within 6 hours and MODE 4 within 12 hours. This changes the CTS by separating out conditions when the LCO is not met and provides additional time to restore the required pressurizer heaters before a plant shutdown is required.

The purpose of the CTS actions is to provide appropriate remedial actions to ensure continued safe operation until the LCO can be restored. This change is acceptable because the proposed ITS Completion Time to restore the required pressurizer heaters to OPERABLE status has been shown to be acceptable based on the infrequent use of the Required Action and the small incremental effect on plant risk as discussed in WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010. The ITS action is also consistent with the equivalent ISTS condition with no pressurizer heater groups OPERABLE. This change is designated as less restrictive because more time will be allowed to complete a Required Action in the ITS than is allowed in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

3.4.4

LCO 3.4.9 The pressurizer shall be OPERABLE with:

≤ 68%

a. Pressurizer water level ~~< [60]~~% and 2

a

b. ~~Two groups of~~ pressurizer heaters OPERABLE with ~~the~~ capacity ~~[of each group]~~ ≥ ~~[~~150~~]~~ kW ~~[~~and capable of being powered from an emergency power supply~~]~~. 2

Applicability M01

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.4 Action. A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open.	6 hours
	AND	
M01	A.2 Be in MODE 4.	~~[~~12~~]~~ hours 2
~~B. One required group of pressurizer heaters inoperable.~~	~~B.1 Restore required group of pressurizer heaters to OPERABLE status.~~	~~72 hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B ~~C.~~ ~~NOTE~~ ~~Not applicable when second group of required pressurizer heaters intentionally made inoperable.~~ 3.4.4 Action ~~Two~~ required ~~groups of~~ pressurizer heaters inoperable.	B ~~C~~.1 Restore ~~at least one group of~~ required pressurizer heaters to OPERABLE status.	24 hours
C ~~D~~. Required Action and associated Completion Time of Condition B ~~or C~~ not met.	C ~~D~~.1 Be in MODE 3. AND C ~~D~~.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.9.1 Verify pressurizer water level is ≤ 68% ~~< [60]~~%.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~REVIEWER'S NOTE~~ ~~The frequency for performing pressurizer heater capacity testing shall be either 18 months or 92 days, depending on whether or not the plant has dedicated safety-related heaters. For dedicated safety-related heaters, which do not normally operate, 92 days is applied. For non-dedicated safety-related heaters, which normally operate, 18 months is applied.~~		
M02 SR 3.4.9.2	Verify capacity of ~~each~~ required ~~group of~~ pressurizer heaters ≥ ~~[~~150~~]~~ kW.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
M03 SR 3.4.9.3	~~[~~ Verify required pressurizer heaters are capable of being powered from an emergency power supply.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

3.4.3

LCO 3.4.9 The pressurizer shall be OPERABLE with:

≤ 68%

a. Pressurizer water level ~~< [60]~~% and 2

b. Two groups of pressurizer heaters OPERABLE with the capacity ~~[~~of each group~~]~~ ≥ ~~[~~150~~]~~ kW ~~[~~and capable of being powered from an emergency power supply~~]~~. 2

Applicability

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.3 Action c. A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open. AND A.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours 2
3.4.3 Action a. B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.3 Action b. Note C. ---------- NOTE ---------- Not applicable when second group of required pressurizer heaters intentionally made inoperable. ---------- 3.4.3 Action b. Two required groups of pressurizer heaters inoperable.	C.1 Restore at least one group of required pressurizer heaters to OPERABLE status.	24 hours
3.4.3 Action a., b. D. Required Action and associated Completion Time of Condition B or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours (2)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.4.3.1 SR 3.4.9.1 Verify pressurizer water level is ~~< [60]~~ ≤ 68%.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ (2)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
~~REVIEWER'S NOTE~~ ~~The frequency for performing pressurizer heater capacity testing shall be either 18 months or 92 days, depending on whether or not the plant has dedicated safety-related heaters. For dedicated safety-related heaters, which do not normally operate, 92 days is applied. For non-dedicated safety-related heaters, which normally operate, 18 months is applied.~~			2
4.4.3.2 SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters ≥ ~~[~~150~~]~~ kW.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.4.3.3 SR 3.4.9.3	~~[~~ Verify required pressurizer heaters are capable of being powered from an emergency power supply.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~	2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.9, PRESSURIZER

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters and their backup heater controls, and emergency power supplies. Pressurizer safety valves and ~~pressurizer~~ power operated relief valves (PORVs) are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "~~Pressurizer~~ Power Operated Relief Valve~~s~~ (PORV~~s~~)," respectively.

block valves

Block Valves

1

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control, using the sprays and heaters during normal operation and proper pressure response for anticipated design basis transients. The water level limit serves two purposes:

a. Pressure control during normal operation maintains subcooled reactor coolant in the loops and thus in the preferred state for heat transport and

b. By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes that could result in degraded ability for pressure control.

The maximum water level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus, both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus ensuring that pressure relief devices (PORVs or pressurizer safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the Safety Limit of 2750 ~~psig~~.

psia

1

1

BASES

BACKGROUND (continued)

The requirement to have ~~two groups~~ of pressurizer heaters ≥ 150 kw capacity ensures that RCS pressure can be maintained. The pressurizer heaters maintain RCS pressure to keep the reactor coolant subcooled. Inability to control RCS pressure during natural circulation flow could result in loss of single phase flow and decreased capability to remove core decay heat. 1

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No safety analyses are performed in lower MODES. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the U FSAR do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. 1

Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 1), is the reason for their inclusion. The requirement for emergency power supplies is based on NUREG-0737 (Ref. 1). The intent is to keep the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power. While loss of offsite power is a coincident occurrence assumed in the accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated in the accident analyses.

The pressurizer satisfies Criterion 2 and Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

~~REVIEWER'S NOTE~~
~~Plants licensed prior to the issuance of NUREG-0737 may not have a requirement on the number of pressurizer groups.~~ 2

The LCO requirement for the pressurizer to be OPERABLE with water level ~~< [60]~~ ≤ 68 % ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. 2

BASES

LCO (continued)

The LCO requires ~~two groups of~~ OPERABLE pressurizer heaters~~, [each]~~ with a capacity ≥ ~~[~~150~~]~~ kW ~~[~~and capable of being powered from an emergency power supply~~]~~. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide subcooling margin to saturation can be obtained in the loops. The exact design value of ~~[~~150~~]~~ kW is derived from the use of 12 heaters rated at 12.5 kW each. The amount needed to maintain pressure is dependent on the ambient heat losses.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, Applicability has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup. ~~The LCO does not apply to MODE 5 (Loops Filled) because LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," applies. The LCO does not apply to MODES 5 and 6 with partial loop operation.~~

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES gives the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Shutdown Cooling System is in service and therefore the LCO is not applicable.

ACTIONS

A.1 and A.2

With pressurizer water level not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within ~~[~~12~~]~~ hours. This takes the plant out of the applicable MODES and restores the plant to operation within the bounds of the safety analyses.

BASES

ACTIONS (continued)

Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Further pressure and temperature reduction to MODE 4 brings the plant to a MODE where the LCO is not applicable. The 12 hour time to reach the nonapplicable MODE is reasonable based on operating experience for that evolution.

~~B.1~~

~~If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours [or in accordance with the Risk Informed Completion Time Program]. The Completion Time of 72 hours is reasonable considering that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station powered heaters.~~ 4

B
~~C~~.1

If ~~two~~ the required ~~groups of~~ pressurizer heaters are inoperable, restoring ~~at least one group of~~ the pressurizer heaters to OPERABLE status is required within 24 hours. ~~The Condition is modified by a Note stating it is not applicable if the second group of required pressurized heaters is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one group of required pressurized heaters is inoperable for any reason and the second group of required pressurized heaters is discovered to be inoperable, or if both groups of required pressurized heaters are discovered to be inoperable at the same time.~~ 4 If ~~both~~ the required ~~groups of~~ pressurizer heaters are inoperable, the pressurizer heaters may not be available to help maintain subcooling in the RCS loops during a natural circulation cooldown following a loss of offsite power. The inoperability of the ~~two groups of~~ required pressurizer heaters during the 24 hour Completion Time has been shown to be acceptable based on the infrequent use of the Required Action and the small incremental effect on plant risk (Ref. 2).

C C 4
~~D~~.1 and ~~D~~.2

If ~~one or more~~ the required ~~group of~~ pressurizer heaters ~~is~~ are inoperable and cannot be restored within the allowed Completion Times, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within ~~[~~12~~]~~ hours. The Completion Time of 6 hours is 2

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BASES

ACTIONS (continued)

reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of ~~[~~12~~]~~ hours is reasonable, based on operating experience, to reach MODE 4 from full power in an orderly manner and without challenging plant systems. 2

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This Surveillance ensures that during steady state operation, pressurizer water level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. ~~[The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

SR 3.4.9.2

~~REVIEWER'S NOTE~~

~~The frequency for performing pressurizer heater capacity testing shall be either 18 months or 92 days, depending on whether or not the plant has dedicated safety-related heaters. For dedicated safety-related heaters, which do not normally operate, 92 days is applied. For non-dedicated safety-related heaters, which normally operate, 18 months is applied.~~ 2

The Surveillance is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) ~~[The Frequency of [18] months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.~~ 2

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

~~[~~ SR 3.4.9.3

~~This SR is not applicable if the heaters are permanently powered by 1E power supplies.~~ 2

This Surveillance demonstrates that the heaters can be manually transferred to and energized by emergency power supplies. ~~[The Frequency of [18] months is based on a typical fuel cycle and industry accepted practice. This is consistent with similar verifications of emergency power.~~ 2

following an actual or simulated loss of offsite power concurrent with a safety injection actuation signal. 1

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]]~~

REFERENCES

1. NUREG-0737, November 1980.

2. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters and their backup heater controls, and emergency power supplies. Pressurizer safety valves and ~~pressurizer~~ power operated relief valves (PORVs) are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "~~Pressurizer~~ Power Operated Relief Valves (PORV~~s~~)," respectively.

block valves

Block Valves

1

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control, using the sprays and heaters during normal operation and proper pressure response for anticipated design basis transients. The water level limit serves two purposes:

a. Pressure control during normal operation maintains subcooled reactor coolant in the loops and thus in the preferred state for heat transport and

b. By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes that could result in degraded ability for pressure control.

The maximum water level limit permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, thus, both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) for anticipated design basis transients, thus ensuring that pressure relief devices (PORVs or pressurizer safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the Safety Limit of 2750 ~~psig~~.

psia

1

1

BASES

BACKGROUND (continued)

The requirement to have two groups of pressurizer heaters ensures that RCS pressure can be maintained. The pressurizer heaters maintain RCS pressure to keep the reactor coolant subcooled. Inability to control RCS pressure during natural circulation flow could result in loss of single phase flow and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No safety analyses are performed in lower MODES. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the U FSAR do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. 1

Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 1), is the reason for their inclusion. The requirement for emergency power supplies is based on NUREG-0737 (Ref. 1). The intent is to keep the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an undefined, but extended, time period after a loss of offsite power. While loss of offsite power is a coincident occurrence assumed in the accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated in the accident analyses.

The pressurizer satisfies Criterion 2 and Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

~~REVIEWER'S NOTE~~
~~Plants licensed prior to the issuance of NUREG-0737 may not have a requirement on the number of pressurizer groups.~~ 2

The LCO requirement for the pressurizer to be OPERABLE with water level ~~< [60]~~ ≤ 68 % ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. 2

BASES

LCO (continued)

The LCO requires two groups of OPERABLE pressurizer heaters, ~~[~~each~~]~~ with a capacity ≥ ~~[~~150~~]~~ kW ~~[~~and capable of being powered from an emergency power supply~~]~~. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide subcooling margin to saturation can be obtained in the loops. The exact design value of ~~[~~150~~]~~ kW is derived from the use of ~~12~~ heaters rated at ~~12.5~~ kW each. The amount needed to maintain pressure is dependent on the ambient heat losses.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, Applicability has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup. ~~The LCO does not apply to MODE 5 (Loops Filled) because LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," applies. The LCO does not apply to MODES 5 and 6 with partial loop operation.~~

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES gives the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Shutdown Cooling System is in service and therefore the LCO is not applicable.

ACTIONS

A.1 and A.2

With pressurizer water level not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within ~~[~~12~~]~~ hours. This takes the plant out of the applicable MODES and restores the plant to operation within the bounds of the safety analyses.

BASES

ACTIONS (continued)

Six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Further pressure and temperature reduction to MODE 4 brings the plant to a MODE where the LCO is not applicable. The 12 hour time to reach the nonapplicable MODE is reasonable based on operating experience for that evolution.

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours ~~[or in accordance with the Risk Informed Completion Time Program]~~. The Completion Time of 72 hours is reasonable considering that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station powered heaters. 2

C.1

If two required groups of pressurizer heaters are inoperable, restoring at least one group of pressurizer heaters to OPERABLE status is required within 24 hours. The Condition is modified by a Note stating it is not applicable if the second group of required pressurized heaters is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one group of required pressurized heaters is inoperable for any reason and the second group of required pressurized heaters is discovered to be inoperable, or if both groups of required pressurized heaters are discovered to be inoperable at the same time. If both required groups of pressurizer heaters are inoperable, the pressurizer heaters may not be available to help maintain subcooling in the RCS loops during a natural circulation cooldown following a loss of offsite power. The inoperability of two groups of required pressurizer heaters during the 24 hour Completion Time has been shown to be acceptable based on the infrequent use of the Required Action and the small incremental effect on plant risk (Ref. 2).

D.1 and D.2

If one or more required group of pressurizer heaters is inoperable and cannot be restored within the allowed Completion Times, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within ~~[~~12~~]~~ hours. The Completion Time of 6 hours is 2

2

1

BASES

ACTIONS (continued)

reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of ~~[~~12~~]~~ hours is reasonable, based on operating experience, to reach MODE 4 from full power in an orderly manner and without challenging plant systems. 2

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This Surveillance ensures that during steady state operation, pressurizer water level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. ~~[The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

SR 3.4.9.2

~~REVIEWER'S NOTE~~

~~The frequency for performing pressurizer heater capacity testing shall be either 18 months or 92 days, depending on whether or not the plant has dedicated safety-related heaters. For dedicated safety-related heaters, which do not normally operate, 92 days is applied. For non-dedicated safety-related heaters, which normally operate, 18 months is applied.~~ 2

The Surveillance is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) ~~[The Frequency of [18] months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.~~ 2

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

~~[~~ SR 3.4.9.3

~~This SR is not applicable if the heaters are permanently powered by 1E power supplies.~~ 2

This Surveillance demonstrates that the heaters can be manually transferred to and energized by emergency power supplies. ~~[The Frequency of [18] months is based on a typical fuel cycle and industry accepted practice. This is consistent with similar verifications of emergency power.~~ 2

following an actual or simulated loss of offsite power concurrent with a safety injection actuation signal. 1

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]]~~

REFERENCES

1. NUREG-0737, November 1980.

2. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.9, BASES, PRESSURIZER

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The ISTS Applicability discussion related to when the LCO does not apply in MODES 5 and 6 is unnecessary detail. Therefore, this information is not included in the St. Lucie Unit 1 and Unit 2 ITS Bases.

4. Changes have been made to be consistent with changes made to the Specifications.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.9, PRESSURIZER

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 10

3.4.10, Pressurizer Safety Valves

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.4.10

~~3.4.3~~ ~~All~~ Three pressurizer ~~code~~ safety valves shall be OPERABLE with ~~a~~ lift setting ~~of~~ s ≥ 2422.8 psig and ≤ 2560.3 psig.

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures > 281°F.

Add proposed 3.4.10 Applicability Note — L01

ACTION:

ACTION A

~~a.~~ With one pressurizer ~~code~~ safety valve inoperable, either restore ~~the inoperable~~ valve to OPERABLE status within 15 minutes or be in ~~HOT~~ 12

ACTION B

MODE 3 ~~STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ within ~~the next~~ ~~6~~ hours.

MODE 4 with any RCS RCS cold leg temperature ≤ 281°F — M01

ACTION B

~~b.~~ ~~With~~ two or more pressurizer ~~code~~ safety valves inoperable, be in ~~HOT STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ with ~~all~~ RCS cold leg temperatures ≤ 281°F within ~~the next~~ ~~6~~ hours.

MODE 3 — MODE 4 — any — 12

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

~~4.4.3~~ Verify each pressurizer ~~code~~ safety valve~~s~~ is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, ~~as left~~ lift settings shall be within +/- 1% ~~of 2500 psia~~.

LA01

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

~~**NOTE**~~

~~The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure~~.

LA02

Three

s

LCO 3.4.10

~~3.4.2.2~~ ~~All~~ pressurizer ~~code~~ safety valves shall be OPERABLE with ~~a~~ lift setting of ≥ 2410.3 psig and ≤ 2560.3 psig.

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures > 230°F.

Add proposed 3.4.10 Applicability Note

L01

ACTION:

ACTION A

~~a.~~ With one pressurizer ~~code~~ safety valve inoperable, either restore ~~the inoperable~~ valve to OPERABLE status within 15 minutes or be in ~~HOT~~ 12

ACTION B

MODE 3 ~~STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ within ~~the next~~ ~~6~~ hours.

MODE 4 with any RCS RCS cold leg temperature ≤ 230°F

M01

ACTION B

~~b.~~ ~~With t~~two or more pressurizer ~~code~~ safety valves inoperable, be in ~~HOT STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ with ~~all~~ RCS cold leg temperatures at ≤ 230°F within ~~the next~~ ~~6~~ hours.

MODE 3

MODE 4

any

12

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

~~4.4.2.2~~ Verify each pressurizer ~~code~~ safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, ~~as-left~~ lift settings shall be within +/- 1% ~~of 2500 psia~~.

LA01

DISCUSSION OF CHANGES
ITS 3.4.10, PRESSURIZER SAFETY VALVES

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.4.3 and Unit 2 CTS 3.4.2.2 Action a. requires, with one pressurizer safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT STANDBY (i.e., MODE 3) within 6 hours and in HOT SHUTDOWN (i.e., MODE 4) within the next 6 hours.

ITS 3.4.10 ACTION A states, with one pressurizer safety valve inoperable, the valve must be restored to OPERABLE status within 15 minutes. If this cannot be met, ITS 3.4.10 ACTION B requires the unit to be in MODE 3 in 6 hours and MODE 4 with any RCS cold leg temperature < 281°F (Unit 1) and < 230°F (Unit 2) in 12 hours. This changes the CTS by requiring the unit, in addition to transitioning to MODE 4, to also reduce RCS cold leg temperature < 281°F (Unit 1) and < 230°F (Unit 2) when only one pressurizer safety valve is inoperable and cannot be restored to OPERABLE status.

The purpose of Unit 1 CTS 3.4.3 and Unit 2 CTS 3.4.2.2 actions is to provide requirements on pressurizer safety valves during MODES 1, 2, 3, and MODE 4 with all RCS cold leg temperatures > 281°F (Unit 1) and > 230°F (Unit 2), and if the Required Action cannot be met within the required Completion Time, the plant must be brought to a MODE in which the requirement does not apply. This change is designated as more restrictive as it provides an explicit RCS cold leg temperature that must be met in MODE 4 to exit the MODE of Applicability when one safety valve is inoperable and cannot be restored to OPERABLE status.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 4.4.3 and Unit 2 CTS 4.4.2.2 require verification that each pressurizer code safety valve is OPERABLE with an additional statement that following testing, as left lift settings shall be within ± 1% of 2500 psia. ITS SR 3.4.10.1 provides a similar requirement. However, the design pressure of

2500 psia is moved to the ITS Bases. This changes the CTS by moving the RCS system design pressure from the CTS to the ITS Bases.

The purpose of the Surveillance is to ensure the pressurizer safety valves setpoints are within ± 1% of their setpoint, which corresponds to the RCS system design pressure of 2500 psia. The removal of the system design pressure from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for the pressurizer safety valves to be verified OPERABLE in accordance with the INSERVICE TESTING PROGRM and that following testing, the lift settings shall be within ± 1%. The RCS system design pressure corresponding to the lift setting is described in the Bases. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 **Unit 2 only:** *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.4.2.2 is modified by a note that states that the pressurizer safety valves lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. This information is not provided in ITS 3.4.10. This changes the CTS by moving this information to the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.4.10 still retains a requirement for the pressurizer safety valves to be OPERABLE. Under the definition of OPERABILITY, the pressurizer safety valves must be capable of lifting at the assumed conditions, which includes the ambient operating conditions of the pressurizer safety valves themselves. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being moved from the Technical Specifications to the ITS Bases.

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* Unit 1 CTS 3.4.3 and Unit 2 CTS 3.4.2.2, in part, provide requirements for the pressurizer code safety valves. ITS LCO 3.4.10 Applicability is modified by a Note that allows the lift settings to not be within the LCO limits during MODES 3 and 4 for the purpose of (in-situ) setting of the pressurizer safety valves under ambient (hot) conditions. The exception is allowed for 54 hours following entry into MODE 3 provided a

preliminary cold setting was made prior to heatup. This changes the CTS by allowing entry into MODES 3 and 4 without verifying that the pressurizer code safety valve lift settings are within the LCO limits.

The purpose of the Applicability Note is to allow entry into MODE 3 to perform testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. This change is acceptable because the requirements continue to ensure that the components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. The cold lift settings give assurance that the valves are OPERABLE near their design condition during the short period of time allowed to verify the settings at the hot condition. While PSL does not set pressurizer safety valves while installed at this time, this Applicability Note provides the flexibility to utilize this method in the future. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

3.4.3 LCO 3.4.10 Three ~~[Two]~~ pressurizer safety valves shall be OPERABLE with lift settings ≥ ~~[2475] psia~~ 2422.8 psig and ≤ ~~[2525] psia~~ 2560.3 psig. (2)

Applicability APPLICABILITY: MODES 1, 2, and 3, MODE 4 with all RCS cold leg temperatures > 281°F ~~greater than the LTOP enable temperature specified in the PTLR~~. (1)

DOC L01

----------NOTE----------
The lift settings are not required to be within LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for ~~[36]~~ 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup. (1)

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.	A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
Action a., b.	B. Required Action and associated Completion Time not met. OR	B.1 Be in MODE 3. AND	6 hours
Action b.		B.2 Be in MODE 4 with any RCS cold leg temperature ≤ 281°F ~~less than or equal to the LTOP enable temperature specified in the PTLR~~.	12 ~~[24]~~ hours (2)
Action b.	Two ~~[~~or more~~]~~ pressurizer safety valves inoperable.		(1)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.4.3	SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

3.4.2.2 LCO 3.4.10 Three ~~[Two]~~ pressurizer safety valves shall be OPERABLE with lift settings ≥ ~~[2475] psia~~ 2410.3 psig and ≤ ~~[2525] psia~~ 2560.3 psig. (2)

Applicability APPLICABILITY: MODES 1, 2, and 3, MODE 4 with all RCS cold leg temperatures > 230°F ~~greater than the LTOP enable temperature specified in the PTLR~~. (1)

DOC L01

----------NOTE----------
The lift settings are not required to be within LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for ~~[36]~~ 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup. (1)

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.	A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
Action a., b.	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
		AND	
Action b.	OR	B.2 Be in MODE 4 with any RCS cold leg temperature ≤ 230°F ~~less than or equal to the LTOP enable temperature specified in the PTLR~~.	12 ~~[24]~~ hours (2)
Action b.	Two ~~[~~or more~~]~~ pressurizer safety valves inoperable.		(1)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.4.2.2	SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.10, PRESSURIZER SAFETY VALVES

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

three The purpose of the ~~two~~ spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the

three Reactor Protection System, ~~two~~ valves are used to ensure that the Safety Limit (SL) of 2750 psia is not exceeded for analyzed transients during

Three operation in MODES 1 and 2. ~~Two~~ safety valves are used for MODE 3 and portions of MODE 4. For the remainder of MODE 4, MODE 5, and MODE 6 with the head on, overpressure protection is provided by operating procedures and the LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." 1

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME, Boiler and Pressure Vessel Code, Section III (Ref. 1). The required lift pressure is 2500 psia ± 1%. The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The upper and lower pressure limits are based on the ± 1%-tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

U All accident analyses in the FSAR that require safety valve actuation

three assume operation of ~~both~~ pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also

all based on operation of ~~both~~ safety valves and assumes that the valves open at the high range of the setting (2500-psia system design pressure plus 1%). These valves must accommodate pressurizer insurges that 1 1 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

could occur during a startup, rod withdrawal, ejected rod, loss of main feedwater, or main feedwater line break accident. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this specification is required to ensure that the accident analysis and design basis calculations remain valid.

The pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

three

The ~~[two]~~ pressurizer safety valves are set to open at the RCS design pressure (2500 psia) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions, and to comply with ASME Code requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or

more

~~both~~ valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above ~~the LTOP~~ ~~temperature~~, OPERABILITY of ~~[two]~~ valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require ~~both~~ safety valves for protection.

281°F

three

three

The LCO is not applicable in MODE 4 when any RCS cold leg temperature is ~~less than or equal to the LTOP enable temperature specified in the PTLR and~~ MODE 5 because LTOP protection is provided. ~~Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.~~

≤ 281°F,

, and MODE 6 when the reactor vessel head is on

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The ~~[36]~~ hour exception is based on 18 hour outage time for each of the ~~two~~ valves. The 18 hour period is derived from operating experience that hot testing can be performed within this timeframe.

54

BASES

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperature ~~less than or equal to the LTOP enable temperature specified in the PTLR~~ within ~~[24~~] hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power without challenging plant systems. Similarly, the ~~[24]~~ hours allowed is reasonable, based on operating experience, to reach MODE 4 without challenging plant systems. With any RCS cold leg temperature ~~less than or equal to the LTOP enable temperature specified in the PTLR~~, overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by ~~[two]~~ pressurizer safety valves.

≤ 281°F (1)
2
12
12
2
≤ 281°F (1)
2

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the INSERVICE TESTING PROGRAM. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 1), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ~~± [3]%~~ for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

-2.5% / +3% (2)

REFERENCES

1. ASME Code for Operation and Maintenance of Nuclear Power Plants.

2

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

three The purpose of the ~~two~~ spring loaded pressurizer safety valves is to provide RCS overpressure protection. Operating in conjunction with the three Reactor Protection System, ~~two~~ valves are used to ensure that the Safety Limit (SL) of 2750 psia is not exceeded for analyzed transients during Three operation in MODES 1 and 2. ~~Two~~ safety valves are used for MODE 3 and portions of MODE 4. For the remainder of MODE 4, MODE 5, and MODE 6 with the head on, overpressure protection is provided by operating procedures and the LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." 1

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME, Boiler and Pressure Vessel Code, Section III (Ref. 1). The required lift pressure is 2500 psia ± 1%. The safety valves discharge steam from the pressurizer to a quench tank located in the containment. The discharge flow is indicated by an increase in temperature downstream of the safety valves and by an increase in the quench tank temperature and level.

The upper and lower pressure limits are based on the ± 1%-tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

U All accident analyses in the FSAR that require safety valve actuation three assume operation of ~~both~~ pressurizer safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis is also all based on operation of ~~both~~ safety valves and assumes that the valves open at the high range of the setting (2500-psia system design pressure plus 1%). These valves must accommodate pressurizer insurges that 1 1 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

could occur during a startup, rod withdrawal, ejected rod, loss of main feedwater, or main feedwater line break accident. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at < 15% power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this specification is required to ensure that the accident analysis and design basis calculations remain valid.

The pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

three

The ~~[two]~~ pressurizer safety valves are set to open at the RCS design pressure (2500 psia) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions, and to comply with ASME Code requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more ~~both~~ valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

2

1

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above ~~the LTOP temperature~~ 230°F, OPERABILITY of three ~~[two]~~ valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require three ~~both~~ safety valves for protection.

1

2

1

The LCO is not applicable in MODE 4 when any RCS cold leg temperature is ≤ 230°F, ~~less than or equal to the LTOP enable temperature specified in the PTLR and~~ MODE 5 because LTOP protection is provided. ~~Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned.~~ , and MODE 6 when the reactor vessel head is on

1

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 ~~[36]~~ hour exception is based on 18 hour outage time for each of the ~~two~~ valves. The 18 hour period is derived from operating experience that hot testing can be performed within this timeframe.

2

1

1

BASES

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1 and B.2

If the Required Action cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperature ~~less than or equal to the LTOP enable temperature specified in the PTLR~~ within ~~[~~24~~]~~ hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power without challenging plant systems. Similarly, the ~~[~~24~~]~~ hours allowed is reasonable, based on operating experience, to reach MODE 4 without challenging plant systems. With any RCS cold leg temperature ~~less than or equal to the LTOP enable temperature specified in the PTLR~~, overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by ~~[two]~~ pressurizer safety valves.

≤ 230°F 1
12 2
12 2
≤ 230°F 1
2

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the INSERVICE TESTING PROGRAM. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 1), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± ~~[~~3~~]~~% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

2

REFERENCES

1. ASME Code for Operation and Maintenance of Nuclear Power Plants.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.10, BASES, PRESSURIZER SAFETY VALVES

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.10, PRESSURIZER SAFETY VALVES

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 11

3.4.11, Power Operated Relief Valve (PORV) Block Valves

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTOR COOLANT SYSTEM

PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

LCO 3.4.11

~~3.4.12~~ Each ~~Power Operated Relief Valve (~~PORV~~)~~ Block Valve shall be OPERABLE.

Applicability

APPLICABILITY: MODES 1, 2, and 3.

Insert proposed ITS 3.4.11 ACTIONS Note

L01

ACTION:

PORV

PORV

ACTION A

With one or more block valve~~(s)~~ inoperable, within 1 hour or in accordance with the Risk Informed Completion Time Program ~~either restore the block valve(s) to OPERABLE status or~~ close ~~the~~ block valve~~(s)~~ and remove power from ~~the~~ block valve~~(s)~~; otherwise, be in ~~at least~~

ACTION B

~~HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

MODE 3

PORV

MODE 4

12

A02

SURVEILLANCE REQUIREMENTS

Insert proposed SR 3.4.11.1 Note 1

L02

Insert proposed SR 3.4.11.1 Note 2

L03

SR 3.4.11.1

~~4.4.12~~ ~~Each block valve shall be demonstrated OPERABLE~~ in accordance with the Surveillance Frequency Control Program by operating ~~the~~ valve ~~through one~~ complete cycle ~~of full travel~~.

each PORV block

a

.

A01

REACTOR COOLANT SYSTEM

3/4.4.4 PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

LCO 3.4.11 ~~3.4.4~~ Each ~~Power Operated Relief Valve (~~PORV~~)~~ Block valve shall be OPERABLE~~.~~ ~~No more than one block valve shall be open at any one time~~.

and one PORV block valve shall be closed.

Applicability **APPLICABILITY:** MODES 1, 2 and 3.

Insert proposed ITS 3.4.11 ACTIONS Note — L01

ACTION:

PORV PORV

ACTION A ~~a.~~ With one or more block valve~~(s)~~ inoperable, within 1 hour or in accordance with the Risk Informed Completion Time Program ~~either restore the block valve(s) to OPERABLE status or~~ close ~~the~~ block valve~~(s)~~ and remove power from ~~the~~ block valve~~(s)~~; otherwise, be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in~~ ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

PORV

ACTION C

MODE 3

PORV MODE 4 PORV 12 A02

ACTION B ~~b.~~ With both block valves open, close one block valve within 1 hour, otherwise be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in~~ ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

ACTION C

MODE 3 MODE 4 A02

12

SURVEILLANCE REQUIREMENTS

Insert proposed SR 3.4.11.1 Note 2 — L03

SR 3.4.11.1 ~~4.4.4~~ ~~Each block valve shall be demonstrated OPERABLE~~ in accordance with the Surveillance Frequency Control Program by operating ~~the~~ valve ~~through one~~ complete cycle ~~of full travel~~ unless ~~the~~ block valve ~~is~~ closed ~~with power removed in order to meet the requirements of Action a. or b. above~~.

a

SR 3.4.11.1 Note 1

PORV

each PORV block

in accordance with the Required Actions of this LCO

DISCUSSION OF CHANGES
ITS 3.4.11, POWER OPERATED RELIEF VALVE (PORV) BLOCK VALVES

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.4.12 Action and Unit 2 CTS 3.4.4 Action a. state that with one or more block valve(s) inoperable, within 1 hour or in accordance with the Risk Informed Completion Time Program either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY (i.e., MODE 3) within the next 6 hours and in COLD SHUTDOWN (i.e., MODE 5) within the following 30 hours. ITS 3.4.11 ACTION A states that with one or more PORV block valves inoperable, close the PORV block valve and remove power from the PORV block valve within 1 hour or in accordance with the Risk Informed Completion Time Program, otherwise ITS 3.4.11 ACTION B requires the unit be in MODE 3 within 6 hours and MODE 4 within 12 hours. This changes the CTS by changing the requirement to be in COLD SHUTDOWN (i.e., MODE 5) within 36 hours, to be in MODE 4 in 12 hours.

The purpose of Unit 1 CTS 3.4.12 Action and Unit 2 CTS 3.4.4 Action a. is to provide the appropriate compensatory actions for one or more inoperable PORV block valves inoperable. However, if the Required Action and Completion Time are not met, the current action places the unit in MODE 5 within 36 hours, rather than placing the unit outside the MODE of Applicability (i.e., MODE 4) if the Required Action and Completion Time are not met. Therefore, in accordance with CTS 3.0.2 (ITS LCO 3.0.2) when the unit is in MODE 4, the requirements to be in MODE 5 in Unit 1 CTS 3.4.12 Action and Unit 2 CTS 3.4.4 Action a. are no longer required to be completed. The PORV block valves, when in MODE 4, are no longer required to be OPERABLE. Additionally, the Completion Time to exit the MODE of Applicability is reduced to 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

These changes are designated as administrative changes and are acceptable because they maintain the current requirement to place the unit in a condition in which the PORV block valves are no longer required consistent with the requirements of CTS 3.0.2 (ITS LCO 3.0.2) and, therefore, do not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

DISCUSSION OF CHANGES
ITS 3.4.11, POWER OPERATED RELIEF VALVE (PORV) BLOCK VALVES

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.4.12 Action and Unit 2 CTS 3.4.4 Action a. describe the Actions to be taken when PORV block valve(s) are inoperable. ITS 3.4.11 also describes Actions to be taken when PORV block valve(s) are inoperable and contains a statement (ITS 3.4.11 ACTION Note 1) that separate condition entry is allowed for each PORV block valve. This changes the CTS by adding a Note stating that separate condition entry is allowed for each PORV block valve.

The purpose of the Unit 1 CTS 3.4.12 and Unit 2 CTS 3.4.4 Actions are to provide the appropriate compensatory actions for inoperable PORV block valves. This proposed change will allow separate condition entry for each PORV block valve. The Note clarifies that PORV block valves are treated as separate entities, each with separate Completion Times. These changes are acceptable since the proposed Required Actions provide sufficient time to satisfy the Required Actions. Valve inoperabilities are normally found one at a time, not concurrently. Therefore, the actions to close a PORV block valve and remove its power will apply as each valve is found to be inoperable and not at the same time. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 **Unit 1 Only:** *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.4.12 states, in part, that each PORV block valve be demonstrated OPERABLE by operating the valve through one complete cycle of full travel. ITS SR 3.4.11.1 states, in part, that each PORV block valve be demonstrated OPERABLE by performing a complete cycle of each PORV block valve. It is modified by Note 1 that states "Not required to be performed with block valve closed in accordance with the Required Actions of this LCO." This changes the CTS by not requiring the Surveillance Requirement to be performed with the block valve closed in accordance with the Actions.

The purpose of CTS 4.4.12 is to ensure that the PORV block valve is ready to operate. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The Note provides time to perform the Surveillance to verify the PORV block valve can operate through a complete cycle. Without the Note, the Surveillance would not be met immediately after closing and removing power from the PORV block valve. This change is designated as less restrictive

because Surveillances will be performed less frequently under the ITS than under the CTS.

L03 *(Category 7 – Relaxation of Surveillance Frequency)* Unit 1 CTS 4.4.12 and Unit 2 CTS 4.4.4 state, in part, that each PORV block valve be demonstrated OPERABLE by operating the valve through one complete cycle of full travel. ITS SR 3.4.11.1 states, in part, that each PORV block valve be demonstrated OPERABLE by performing a complete cycle of each PORV block valve. It is modified by Note 2 that states "Only required to be performed in MODES 1 and 2." This changes the CTS by not requiring the Surveillance Requirement to be performed in certain MODES.

The purpose of Unit 1 CTS 4.4.12 and Unit 2 CTS 4.4.4 is to perform the complete cycle of each PORV block valve in accordance with the Surveillance Frequency Control Program. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The ITS SR 3.4.11.1 Note 2 states that the PORV block valve Surveillance is only required to be performed in MODES 1 and 2. Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 ~~Pressurizer~~ Power Operated Relief Valve~~s~~ (PORV~~s~~) Block Valves 1

3.4.12 LCO 3.4.11 Each PORV ~~and associated~~ block valve shall be OPERABLE. 1

Applicability APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------

L01 Separate Condition entry is allowed for each PORV ~~and each~~ block valve. 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One or more PORVs inoperable and capable of being manually cycled.~~	~~A.1 Close and maintain power to associated block valve.~~	~~1 hour~~ 1
~~B. One PORV inoperable and not capable of being manually cycled.~~	~~B.1 Close associated block valve.~~ ~~AND~~ ~~B.2 Remove power from associated block valve.~~ ~~AND~~ ~~B.3 Restore PORV to OPERABLE status.~~	~~1 hour~~ ~~1 hour~~ 1 ~~72 hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.12 Action A or more PORV s ~~C~~. One block valve inoperable.	~~C.1~~ ~~Place associated PORV in manual control.~~ ~~AND~~ ~~C.2~~ ~~Restore block valve to OPERABLE status.~~	~~1 hour~~ ~~72 hours~~ ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
3.4.12 Action B ~~D~~. Required Action and associated Completion Time ~~of Condition A, B, or C~~ not met.	B ~~D~~.1 Be in MODE 3. AND B ~~D~~.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours
~~E.~~ ~~---------- NOTE ----------~~ ~~Not applicable when second PORV intentionally made inoperable.~~ ~~----------~~ ~~Two PORVs inoperable and not capable of being manually cycled.~~	A ~~E~~.1 Close ~~associated~~ block valve~~s~~. PORV AND A ~~E~~.2 Remove power from ~~associated~~ block valve~~s~~. PORV ~~AND~~ ~~E.3~~ ~~Verify LCO 3.7.5, "Auxiliary Feedwater System," is met.~~ ~~AND~~ ~~E.4~~ ~~Restore at least one PORV to OPERABLE status.~~	1 hour 1 hour ~~1 hour~~ ~~8 hours~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~F. Required Actions and Associated Completion Times of Condition E not met.~~	~~F.1 Be in MODE 3.~~ ~~AND~~ ~~F.2 Be in MODE 4.~~	~~6 hours~~ ~~[12] hours~~
~~G. ---------- NOTE ---------- Not applicable when second block valve intentionally made inoperable. ----------~~ ~~Two block valves inoperable.~~	~~G.1 Verify LCO 3.7.5, "Auxiliary Feedwater System," is met.~~ ~~AND~~ ~~G.2 Restore at least one block valve to OPERABLE status.~~	~~1 hour~~ ~~8 hours~~
~~H. Required Action and associated Completion Time of Condition G not met.~~	~~H.1 Be in MODE 3.~~ ~~AND~~ ~~H.2 Be in MODE 4.~~	~~6 hours~~ ~~[12] hours~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.4.12 L02 L03 SR 3.4.11.1	----------NOTES------- PORV ---------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. ---------- Perform a complete cycle of each PORV block valve.	~~[[92] days~~ 1 ~~OR~~ 2 In accordance with the Surveillance Frequency Control Program ~~]~~
~~SR 3.4.11.2~~	~~----------NOTE----------~~ ~~Only required to be performed in MODES 1 and 2.~~ ~~----------~~ ~~Perform a complete cycle of each PORV.~~	~~[[18] months~~ 1 ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program~~]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.4.11.3~~	~~[Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems.~~	~~[[18] months]~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~ 1
~~SR 3.4.11.4~~	~~[Verify PORVs and block valve(s) are capable of being powered from an emergency power supply.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~ 1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 ~~Pressurizer~~ Power Operated Relief Valve~~s~~ (PORV~~s~~) Block Valves 1

3.4.4 LCO 3.4.11 Each PORV ~~and associated~~ block valve and one PORV block valve shall be closed shall be OPERABLE. 1

Applicability APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------

L01 Separate Condition entry is allowed for each PORV ~~and each~~ block valve. 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One or more PORVs inoperable and capable of being manually cycled.~~	~~A.1 Close and maintain power to associated block valve.~~	~~1 hour~~ 1
~~B. One PORV inoperable and not capable of being manually cycled.~~	~~B.1 Close associated block valve.~~ ~~AND~~ ~~B.2 Remove power from associated block valve.~~ ~~AND~~ ~~B.3 Restore PORV to OPERABLE status.~~	~~1 hour~~ ~~1 hour~~ 1 ~~72 hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.4 Action a A or more PORV s ~~C~~. One block valve inoperable.	~~C.1~~ ~~Place associated PORV in manual control.~~ ~~AND~~ ~~C.2~~ ~~Restore block valve to OPERABLE status.~~	~~1 hour~~ ~~72 hours~~ ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
3.4.4 Action a 3.4.4 Action b C ~~D~~. Required Action and associated Completion Time of Condition A, ~~B, or C~~ not met. or	C ~~D~~.1 Be in MODE 3. AND C ~~D~~.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours
~~E.~~ ~~---------- NOTE ----------~~ ~~Not applicable when second PORV intentionally made inoperable.~~ ~~----------~~ ~~Two PORVs inoperable and not capable of being manually cycled.~~	~~E~~.1 Close ~~associated~~ block valve~~s~~. PORV A AND A ~~E~~.2 Remove power from ~~associated~~ block valve~~s~~. PORV ~~AND~~ ~~E.3~~ ~~Verify LCO 3.7.5, "Auxiliary Feedwater System," is met.~~ ~~AND~~ ~~E.4~~ ~~Restore at least one PORV to OPERABLE status.~~	1 hour 1 hour ~~1 hour~~ ~~8 hours~~

3.4.4 Action b

B. Both PORV block valves open.	B.1 Close one PORV block valve.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~F. Required Actions and Associated Completion Times of Condition E not met.~~	~~F.1 Be in MODE 3.~~ ~~AND~~ ~~F.2 Be in MODE 4.~~	~~6 hours~~ ~~[12] hours~~
~~G. ---------- NOTE ---------- Not applicable when second block valve intentionally made inoperable. ----------~~ ~~Two block valves inoperable.~~	~~G.1 Verify LCO 3.7.5, "Auxiliary Feedwater System," is met.~~ ~~AND~~ ~~G.2 Restore at least one block valve to OPERABLE status.~~	~~1 hour~~ ~~8 hours~~
~~H. Required Action and associated Completion Time of Condition G not met.~~	~~H.1 Be in MODE 3.~~ ~~AND~~ ~~H.2 Be in MODE 4.~~	~~6 hours~~ ~~[12] hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	----------NOTES---------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. ---------- Perform a complete cycle of each block valve.	~~[~~ ~~[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
~~SR 3.4.11.2~~	~~----------NOTE----------~~ ~~Only required to be performed in MODES 1 and 2.~~ ~~----------~~ ~~Perform a complete cycle of each PORV.~~	~~[~~ ~~[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program~~]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.4.11.3~~	~~[Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems.~~	~~[[18] months]~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~ 1
~~SR 3.4.11.4~~	~~[Verify PORVs and block valve(s) are capable of being powered from an emergency power supply.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~ 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.11, POWER OPERATED RELIEF VALVE (PORV) BLOCK VALVES

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 ~~Pressurizer~~ Power Operated Relief Valve~~s~~ (PORV~~s~~)

Block Valves

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORV is ~~an air-operated valve that is~~ automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. ~~The PORV may also be manually operated using controls installed in the control room.~~

An electric, motor operated, normally open, block valve is installed between the pressurizer and the PORV. The function of the block valve is to isolate the PORV. Block valve closure is accomplished manually using controls in the control room ~~and may be used to isolate a leaking PORV to permit continued power operation~~. Most importantly, the block ~~valve is~~ used to isolate ~~a stuck open PORV to isolate the resulting small break loss of coolant accident (LOCA)~~. Closure terminates the RCS depressurization and coolant inventory loss.

valves are

an inadvertent opening of one or both PORVs.

The ~~PORV and its~~ block valve controls are powered from normal power supplies. Their controls are also capable of being powered from emergency supplies. ~~Power supplies for the PORV are separate from those for the block valve.~~ Power supply requirements are defined in NUREG-0737, Paragraph II, G.1 (Ref. 1).

~~The PORV setpoint is above the high pressure reactor trip setpoint and below the opening setpoint for the pressurizer safety valves as required by Reference 2. The purpose of the relationship of these setpoints is to limit the number of transient pressure increase challenges that might open the PORV, which, if opened, could fail in the open position. The PORV setpoint thus limits the frequency of challenges from transients and limits the possibility of a small break LOCA from a failed open PORV. Placing the setpoint below the pressurizer safety valve opening setpoint reduces the frequency of challenges to the safety valves, which, unlike the PORV, cannot be isolated if they were to fail to open.~~

The primary purpose of this LCO is to ensure that the PORV ~~and the~~ block valve~~s~~ are operating correctly so the ~~potential for a small break LOCA through the PORV pathway is minimized, or if a small break LOCA were to occur through a failed open PORV, the~~ block valve could be manually operated to isolate the ~~path~~.

s

pathway from an inadvertent opening of one or both PORVs. The PORV block valves are maintained open during power operation to ensure the PORV function is available.

BASES

BACKGROUND (continued)

~~The PORV may be manually operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available, a condition that may be encountered during loss of offsite power. Operators can manually open the PORVs to reduce RCS pressure in the event of a steam generator tube rupture (SGTR) with offsite power unavailable.~~ 3

~~The PORV may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.~~ 3

The PORV functions as an automatic overpressure device and limits challenges to the safety valves. Although the PORV acts as an overpressure device for operational purposes, safety analyses ~~[~~do not take credit for PORV actuation~~, but] do take credit for the safety valves~~. 2

The PORV also provides low temperature overpressure protection (LTOP) during heatup and cooldown. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," addresses this function.

APPLICABLE SAFETY ANALYSES

Insert 1

~~The PORV small break LOCA break size is bounded by the spectrum of piping breaks analyzed for plant licensing. Because the PORV small break LOCA is located at the top of the pressurizer, the RCS response characteristics are different from RCS loop piping breaks; analyses have been performed to investigate these characteristics.~~ 1

~~The possibility of a small break LOCA through the PORV is reduced when the PORV flow path is OPERABLE and the PORV opening setpoint is established to be reasonably remote from expected transient challenges. The possibility is minimized if the flow path is isolated.~~

~~The PORV opening setpoint has been established in accordance with Reference 2. It has been set so expected RCS pressure increases from anticipated transients will not challenge the PORV, minimizing the possibility of small break LOCA through the PORV.~~

~~Overpressure protection is provided by safety valves, and analyses do not take credit for the PORV opening for accident mitigation.~~

block valves

~~Pressurizer~~ PORV~~s~~ satisfy Criterion ~~3~~ 4 of 10 CFR 50.36(c)(2)(ii). 1

1

INSERT 1

The PORV block valves are not credited to mitigate any design basis accident or transient specified in Reference 2. NRC Regulatory Issue Summary 2005-29 (Ref. 3) describes limiting the probability of initiating a more safety significant event as a result of an anticipated transient. The inadvertent PORV opening is an anticipated operational occurrence (AOO) and could result in a pressurizer overfill condition. Such a condition could lead to an inability to isolate the PORV, resulting in a condition similar to a small break LOCA.

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of both pressurizer PORVs. Initially the event results in a loss of RCS fluid and a rapid RCS depressurization. The challenge to the specified acceptable fuel design limits is terminated by the thermal margin / low pressure reactor trip signal, however, the RCS fluid loss and depressurization continue. Safety injection is actuated, and the pressurizer level increases with the potential to result in pressurizer overfill and liquid discharge through the stuck open PORV(s). The minimum time from event initiation to the pressurizer dome becoming liquid filled is 7 minutes. To prevent liquid discharge through the open PORV(s), the operator closes the associated open PORV block valve(s) within 7 minutes prior to the pressurizer dome becoming liquid filled.

1

BASES to terminate a pressurizer overfill event following an inadvertent PORV opening. An OPERABLE block valve may be either closed and energized or open and energized.

LCO

The LCO requires the PORV ~~and its associated~~ block valve to be OPERABLE. The block valve is required to be OPERABLE so it may be used to isolate the flow path ~~if the PORV is not OPERABLE~~.

1

~~Valve OPERABILITY also means the PORV setpoint is correct. By ensuring that the PORV opening setpoint is correct, the PORV is not subject to frequent challenges from possible pressure increase transients, and therefore the possibility of a small break LOCA through a failed open PORV is not a frequent event.~~

3

APPLICABILITY

In MODES 1, 2, and 3, the PORV ~~and its~~ block valve~~s~~ are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. A likely cause for PORV small break LOCA is a result of pressure increase transients that cause the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the steam generators are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODES 1 and 2.

1

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, this LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

The ACTIONS are modified by a Note. The Note clarifies that ~~all~~ ~~pressurizer~~ PORV~~s and~~ block valves are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

1

~~A.1~~

~~With the PORV inoperable and capable of being manually cycled, either the PORV must be restored or the flow path isolated within 1 hour. The block valve should be closed but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. Although the PORV may be designated inoperable, it may be able to be manually opened and closed and in this manner can be used to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for~~

3

1

BASES

ACTIONS (continued)

~~a small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).~~

~~Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that minor problems can be corrected or closure can be accomplished in this time period.~~

~~B.1, B.2, and B.3~~

~~If one PORV is inoperable and not capable of being manually cycled, it must either be isolated, by closing the associated block valve and removing the power from the block valve, or restored to OPERABLE status. The Completion Time of 1 hour is reasonable, based on challenges to the PORVs during this time period, and provides the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time of 1 hour. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~

A A

~~C~~.1 and ~~C~~.2

or more PORV | s are | it must be closed with power removed

If one block valve ~~is~~ inoperable, ~~then~~ it must be restored to OPERABLE status, or ~~the associated PORV placed in manual control~~. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV ~~in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable~~. The Completion Times of 1 hour are reasonable based on the small potential for challenges to the system during this time period and provide the operator time to correct the situation. ~~Because at least one PORV remains OPERABLE, the operator~~

or in accordance with the Risk Informed Completion Time Program

block valve in the closed position with power removed

BASES

ACTIONS (continued)

~~is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status.~~ [Alternatively, ~~a~~ Completion Time can be determined in accordance with the Risk Informed Completion Time Program.] ~~The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B since the PORVs are not capable of automatically mitigating an overpressure event when placed in manual control. If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the PORV restored to OPERABLE status.~~

~~D~~.1 and ~~D~~.2

If the Required Action cannot be met within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~E.1, E.2, E.3, and E.4~~

~~If two PORVs are inoperable and not capable of being manually cycled, it is necessary to isolate the flow path by closing and removing the power to the associated block valves within 1 hour and restore at least one PORV within 8 hours. The Condition is modified by a Note stating it is not applicable if the second PORV train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is applicable if one PORV is inoperable for any reason and the second PORV is discovered to be inoperable, or if both PORVs are discovered to be inoperable at the same time.~~

~~In the event of a loss of feedwater, the PORVs would be used to remove core heat. In order to minimize the consequences of a loss of feedwater while two PORVs are inoperable, Required Action E.3 requires that LCO 3.7.5, "Auxiliary Feedwater System," be met to ensure AFW is available. The inoperability of two PORVs during the 8 hour Completion Time has been shown to be acceptable based on the infrequent use of the Required Action and the small incremental effect on plant risk (Ref. 3). If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two PORVs inoperable.~~

BASES

ACTIONS (continued)

~~F.1 and F.2~~

~~If two PORVs are inoperable and are not capable of being manually cycled and are not restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable, considering that a plant can cool down within that time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.~~

~~G.1 and G.2~~

~~If two block valves are inoperable, it is necessary to restore at least one block valve to OPERABLE status within 8 hours. The Condition is modified by a Note stating it is not applicable if the second block valve is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one block valve is inoperable for any reason and the second block valve is discovered to be inoperable, or if both block valves are discovered to be inoperable at the same time. In the event of a loss of feedwater, the PORVs would be used to remove core heat. In order to minimize the consequences of a loss of feedwater while two block valves are inoperable, Required Action G.2 requires that LCO 3.7.5, "Auxiliary Feedwater System," be verified to be met within 1 hour. The inoperability of two block valves during the 8 hour Completion Time has been shown to be acceptable based on the infrequent use of the Required Actions and the small incremental effect on plant risk (Ref. 3).~~

3

~~H.1 and H.2~~

~~If the Required Actions and associated Completion Times of Condition F or G are not met, then the plant must be brought to a MODE in which the LCO does not apply. The plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3~~

BASES

ACTIONS (continued)

~~from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable considering that a plant can cool down within that time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

PORV

Block valve cycling verifies that it can be closed if necessary. ~~[The basis for the Frequency of [92 days] is the ASME Code (Ref. 4).~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

This SR is modified by two Notes. Note 1 modifies this SR by stating that this SR is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO. ~~Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable.~~ Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. ~~[In accordance with Reference 5, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.]~~

~~SR 3.4.11.2~~

~~SR 3.4.11.2 requires complete cycling of each PORV. PORV cycling demonstrates its function. [The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

~~The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. [In accordance with Reference 4, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.]~~

~~[SR 3.4.11.3~~

~~Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. [The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other surveillances used to demonstrate PORV OPERABILITY.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]]~~

3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[SR 3.4.11.4~~

~~This Surveillance is not required for plants with permanent 1E power supplies to the valves. The test demonstrates that emergency power can be provided and is performed by transferring power from the normal supply to the emergency supply and cycling the valves. [The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]]~~

REFERENCES

1. NUREG-0737, Paragraph II, G.I, November 1980.

~~2. Inspection and Enforcement (IE) Bulletin 79-05B, April 21, 1979.~~

~~3. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.~~

~~4. ASME Code for Operation and Maintenance of Nuclear Power Plants.~~

~~[5. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," June 25, 1990.]~~

2. UFSAR Chapter 15

3. NRC RIS 2005-29, "Anticipated Transients That Could Develop Into More Serious Events," December 14, 2005

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 ~~Pressurizer~~ Power Operated Relief Valve~~s~~ (PORV~~s~~) Block Valves 1

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORV is ~~an air-operated valve that is~~ automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. ~~The PORV may also be manually operated using controls installed in the control room.~~ 1 3

An electric, motor operated, normally open, block valve is installed between the pressurizer and the PORV. The function of the block valve is to isolate the PORV. Block valve closure is accomplished manually using controls in the control room ~~and may be used to isolate a leaking PORV to permit continued power operation~~. Most importantly, the block ~~valve is~~ used to isolate ~~a stuck open PORV to isolate the resulting small break loss of coolant accident (LOCA)~~. Closure terminates the RCS depressurization and coolant inventory loss. 1

valves are

an inadvertent opening of one or both PORVs.

The ~~PORV and its~~ block valve controls are powered from normal power supplies. Their controls are also capable of being powered from emergency supplies. ~~Power supplies for the PORV are separate from those for the block valve.~~ Power supply requirements are defined in NUREG-0737, Paragraph II, G.1 (Ref. 1). 1

~~The PORV setpoint is above the high pressure reactor trip setpoint and below the opening setpoint for the pressurizer safety valves as required by Reference 2. The purpose of the relationship of these setpoints is to limit the number of transient pressure increase challenges that might open the PORV, which, if opened, could fail in the open position. The PORV setpoint thus limits the frequency of challenges from transients and limits the possibility of a small break LOCA from a failed open PORV. Placing the setpoint below the pressurizer safety valve opening setpoint reduces the frequency of challenges to the safety valves, which, unlike the PORV, cannot be isolated if they were to fail to open.~~ 3

The primary purpose of this LCO is to ensure that the PORV ~~and the~~ block valve are operating correctly so the ~~potential for a small break LOCA through the PORV pathway is minimized, or if a small break LOCA were to occur through a failed open PORV, the~~ block valve could be manually operated to isolate the ~~path~~. 1

s

pathway from an inadvertent opening of one or both PORVs. Each PORV capacity is sufficient to avoid lifting the safety valves during power operations. Therefore, one PORV block valve is maintained open during power operation to ensure the PORV function is available. The second PORV block valve is closed to avoid excessive loss of reactor coolant inventory upon PORV actuation.

BASES

BACKGROUND (continued)

~~The PORV may be manually operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available, a condition that may be encountered during loss of offsite power. Operators can manually open the PORVs to reduce RCS pressure in the event of a steam generator tube rupture (SGTR) with offsite power unavailable.~~

~~The PORV may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.~~

The PORV functions as an automatic overpressure device and limits challenges to the safety valves. Although the PORV acts as an overpressure device for operational purposes, safety analyses ~~[~~do not take credit for PORV actuation~~, but] do take credit for the safety valves~~.

The PORV also provides low temperature overpressure protection (LTOP) during heatup and cooldown. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," addresses this function.

APPLICABLE SAFETY ANALYSES

Insert 1

~~The PORV small break LOCA break size is bounded by the spectrum of piping breaks analyzed for plant licensing. Because the PORV small break LOCA is located at the top of the pressurizer, the RCS response characteristics are different from RCS loop piping breaks; analyses have been performed to investigate these characteristics.~~

~~The possibility of a small break LOCA through the PORV is reduced when the PORV flow path is OPERABLE and the PORV opening setpoint is established to be reasonably remote from expected transient challenges. The possibility is minimized if the flow path is isolated.~~

~~The PORV opening setpoint has been established in accordance with Reference 2. It has been set so expected RCS pressure increases from anticipated transients will not challenge the PORV, minimizing the possibility of small break LOCA through the PORV.~~

~~Overpressure protection is provided by safety valves, and analyses do not take credit for the PORV opening for accident mitigation.~~

block valves

~~Pressurizer~~ PORV~~s~~ satisfy Criterion ~~3~~ 4 of 10 CFR 50.36(c)(2)(ii).

1

INSERT 1

The PORV block valves are not credited to mitigate any design basis accident or transient specified in Reference 2. NRC Regulatory Issue Summary 2005-29 (Ref. 3) describes limiting the probability of initiating a more safety significant event as a result of an anticipated transient. The inadvertent PORV opening is an anticipated operational occurrence (AOO) and could result in a pressurizer overfill condition. Such a condition could lead to an inability to isolate the PORV, resulting in a condition similar to a small break LOCA.

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of one pressurizer PORV. Initially the event results in a loss of RCS fluid and a rapid RCS depressurization. The challenge to the specified acceptable fuel design limits is terminated by the thermal margin / low pressure reactor trip signal, however, the RCS fluid loss and depressurization continue. Safety injection is actuated, and the pressurizer level increases with the potential to result in pressurizer overfill and liquid discharge through the stuck open PORV(s). The minimum time from event initiation to the pressurizer dome becoming liquid filled is just under 3 minutes. To prevent liquid discharge through the open PORV(s), the operator closes the associated open PORV block valve(s) within just under 3 minutes prior to the pressurizer dome becoming liquid filled.

1

BASES

to terminate a pressurizer overfill event following an inadvertent PORV opening. An OPERABLE PORV block valve may be either closed and energized or open and energized. Additionally, one PORV block valve must be closed to preclude RCS blowdown from both PORV pathways as a result of inadvertent opening of both PORVs.

LCO

The LCO requires the PORV ~~and its associated~~ block valve to be OPERABLE. The block valve is required to be OPERABLE so it may be used to isolate the flow path ~~if the PORV is not OPERABLE~~.

1

~~Valve OPERABILITY also means the PORV setpoint is correct. By ensuring that the PORV opening setpoint is correct, the PORV is not subject to frequent challenges from possible pressure increase transients, and therefore the possibility of a small break LOCA through a failed open PORV is not a frequent event.~~

3

APPLICABILITY

s

In MODES 1, 2, and 3, the PORV ~~and its~~ block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. A likely cause for PORV small break LOCA is a result of pressure increase transients that cause the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the steam generators are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODES 1 and 2.

1

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, this LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

the

The ACTIONS are modified by a Note. The Note clarifies that ~~all~~ ~~pressurizer~~ PORV~~s and~~ block valves are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

1

~~A.1~~

~~With the PORV inoperable and capable of being manually cycled, either the PORV must be restored or the flow path isolated within 1 hour. The block valve should be closed but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. Although the PORV may be designated inoperable, it may be able to be manually opened and closed and in this manner can be used to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for~~

3

1

BASES

ACTIONS (continued)

~~a small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).~~

~~Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that minor problems can be corrected or closure can be accomplished in this time period.~~

~~B.1, B.2, and B.3~~

~~If one PORV is inoperable and not capable of being manually cycled, it must either be isolated, by closing the associated block valve and removing the power from the block valve, or restored to OPERABLE status. The Completion Time of 1 hour is reasonable, based on challenges to the PORVs during this time period, and provides the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time of 1 hour. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~

A A

~~C~~.1 and ~~C~~.2

or more PORV | s are | it must be closed with power removed

If one block valve ~~is~~ inoperable, ~~then~~ it must be restored to OPERABLE status, or ~~the associated PORV placed in manual control~~. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV ~~in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable~~. The Completion Times of 1 hour are reasonable based on the small potential for challenges to the system during this time period and provide the operator time to correct the situation. ~~Because at least one PORV remains OPERABLE, the operator~~

or in accordance with the Risk Informed Completion Time Program

block valve in the closed position with power removed

BASES

ACTIONS (continued)

~~is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status.~~ ~~[~~Alternatively, ~~a~~ Completion Time s can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ ~~The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B since the PORVs are not capable of automatically mitigating an overpressure event when placed in manual control. If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the PORV restored to OPERABLE status.~~ 3

C C

~~D~~.1 and ~~D~~.2

B.1
If both PORV block valves are open, it is necessary to isolate one PORV flow path by closing one PORV block valve within 1 hour. The Completion Time of 1 hour is reasonable based on the small potential for challenges to the system during this time period and provide the operator time to correct the situation. 1

If the Required Action cannot be met within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~E.1, E.2, E.3, and E.4~~

~~If two PORVs are inoperable and not capable of being manually cycled, it is necessary to isolate the flow path by closing and removing the power to the associated block valves within 1 hour and restore at least one PORV within 8 hours. The Condition is modified by a Note stating it is not applicable if the second PORV train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is applicable if one PORV is inoperable for any reason and the second PORV is discovered to be inoperable, or if both PORVs are discovered to be inoperable at the same time.~~

~~In the event of a loss of feedwater, the PORVs would be used to remove core heat. In order to minimize the consequences of a loss of feedwater while two PORVs are inoperable, Required Action E.3 requires that LCO 3.7.5, "Auxiliary Feedwater System," be met to ensure AFW is available. The inoperability of two PORVs during the 8 hour Completion Time has been shown to be acceptable based on the infrequent use of the Required Action and the small incremental effect on plant risk (Ref. 3). If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two PORVs inoperable.~~ 3

BASES

ACTIONS (continued)

~~F.1 and F.2~~

~~If two PORVs are inoperable and are not capable of being manually cycled and are not restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable, considering that a plant can cool down within that time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.~~

~~G.1 and G.2~~

~~If two block valves are inoperable, it is necessary to restore at least one block valve to OPERABLE status within 8 hours. The Condition is modified by a Note stating it is not applicable if the second block valve is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one block valve is inoperable for any reason and the second block valve is discovered to be inoperable, or if both block valves are discovered to be inoperable at the same time. In the event of a loss of feedwater, the PORVs would be used to remove core heat. In order to minimize the consequences of a loss of feedwater while two block valves are inoperable, Required Action G.2 requires that LCO 3.7.5, "Auxiliary Feedwater System," be verified to be met within 1 hour. The inoperability of two block valves during the 8 hour Completion Time has been shown to be acceptable based on the infrequent use of the Required Actions and the small incremental effect on plant risk (Ref. 3).~~

3

~~H.1 and H.2~~

~~If the Required Actions and associated Completion Times of Condition F or G are not met, then the plant must be brought to a MODE in which the LCO does not apply. The plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3~~

BASES

ACTIONS (continued)

~~from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable considering that a plant can cool down within that time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

PORV

Block valve cycling verifies that it can be closed if necessary. ~~[The basis for the Frequency of [92 days] is the ASME Code (Ref. 4).~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

This SR is modified by two Notes. Note 1 modifies this SR by stating that this SR is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO. ~~Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable.~~ Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. ~~[In accordance with Reference 5, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.]~~

~~SR 3.4.11.2~~

~~SR 3.4.11.2 requires complete cycling of each PORV. PORV cycling demonstrates its function. [The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. [In accordance with Reference 4, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.]~~

~~[SR 3.4.11.3~~

~~Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. [The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other surveillances used to demonstrate PORV OPERABILITY.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]]~~

3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[SR 3.4.11.4~~

~~This Surveillance is not required for plants with permanent 1E power supplies to the valves. The test demonstrates that emergency power can be provided and is performed by transferring power from the normal supply to the emergency supply and cycling the valves. [The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]]~~

REFERENCES

1. NUREG-0737, Paragraph II, G.I, November 1980.

~~2. Inspection and Enforcement (IE) Bulletin 79-05B, April 21, 1979.~~

~~3. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.~~

~~4. ASME Code for Operation and Maintenance of Nuclear Power Plants.~~

~~[5. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," June 25, 1990.]~~

2. UFSAR Chapter 15

3. NRC RIS 2005-29, "Anticipated Transients That Could Develop Into More Serious Events," December 14, 2005

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.11, BASES, POWER OPERATED RELIEF VALVE (PORV) BLOCK VALVES

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the changes to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.11, POWER OPERATED RELIEF VALVE (PORV) BLOCK VALVES

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 12

3.4.12, Low Temperature Overpressure Protection (LTOP) System

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

Low Temperature Overpressure Protection (LTOP) System

A01

REACTOR COOLANT SYSTEM

~~POWER OPERATED RELIEF VALVES~~

Add LCO 3.4.12 high pressure safety injection (HPSI) pump and charging pump requirements

M01

Add LCO 3.4.12 safety injection tanks (SITs) requirements

M02

LIMITING CONDITION FOR OPERATION

lift settings

LCO 3.4.12.a ~~3.4.13~~ Two power operated relief valves (PORVs) shall be OPERABLE, with ~~their setpoints selected to the low temperature mode of operation as follows~~:

≤

LCO 3.4.12.a ~~a.~~ ~~A setpoint of less than or equal to~~ 350 psia ~~shall be selected~~ during heatup, cooldown and isothermal conditions when ~~the temperature of~~ any RCS cold leg ~~is less than or equal to~~ 200°F.

≤ temperature is ≤

LCO 3.4.12.a ~~b.~~ ~~A setpoint of less than or equal to~~ 530 psia ~~shall be selected~~ during heatup, cooldown and isothermal conditions when ~~the temperature of~~ any RCS cold leg ~~is greater than~~ 200°F and ~~less than or equal to~~ 300°F.

temperature is >

≤

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 300°F, MODE 5, and MODE 6 when the head is on the reactor vessel; ~~and the~~ RCS ~~is not vented through~~ greater than a 1.75 square inch ~~vent~~.

LCO 3.4.12.b

depressurized and an RCS vent of ≥

ACTION:

required

required

ACTION E ~~a.~~ ~~With~~ one PORV inoperable in MODE 4, restore ~~the inoperable~~ PORV to OPERABLE status within 7 days; or depressurize ~~and vent the~~ RCS ~~through~~ greater than a 1.75 square inch ~~vent~~ within ~~the next 8~~ hours.

establish

12

L01

ACTION G

and establish RCS vent of ≥

required

required

ACTION F ~~b.~~ ~~With~~ one PORV inoperable in MODES 5 or 6, ~~either (1)~~ restore ~~the inoperable~~ PORV to OPERABLE status within 24 hours, ~~or (2) complete~~ depressurization and venting of the RCS ~~through~~ greater than a 1.75 square inch ~~vent~~ within ~~a total of 32~~ hours.

ACTION G

e

and establish RCS vent of ≥

12

L01

two required

Condition G ~~c.~~ ~~With both~~ PORVs inoperable, ~~restore at least one PORV to operable status or complete~~ depressurization and venting of the RCS ~~through~~ greater than a 1.75 square inch ~~vent~~ within ~~24~~ hours.

ACTION G

12

M03

and establish RCS vent of ≥

e

required RCS vent ≥ 1.75 inches is open

SR 3.4.12.4; 2nd Frequency ~~d.~~ With the RCS vented per ACTIONS a, b, or c, verify ~~the vent pathway at least~~ once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify ~~the vent pathway~~ every 12 hours.

SR 3.4.12.4; 1st Frequency

required RCS vent ≥ 1.75 inches is open

~~e.~~ ~~In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.~~

L02

3.4.12 ACTIONS Note ~~f.~~ LCO 3.0.4.b is not applicable to PORVs when entering MODE 4.

Insert SR 3.4.12.1, SR 3.4.12.2

M01

SURVEILLANCE REQUIREMENTS

Insert SR 3.4.12.3

M02

~~4.4.13~~ Each PORV shall be demonstrated OPERABLE by:

block

for each required PORV

SR 3.4.12.5 ~~a.~~ Verify~~ing the~~ PORV ~~isolation~~ valve is open in accordance with the Surveillance Frequency Control Program; and

on each required PORV

actuation

SR 3.4.12.6 ~~b.~~ Perform~~ance of a~~ CHANNEL FUNCTION TEST, ~~but~~ excluding ~~valve operation~~, in accordance with the Surveillance Frequency Control Program; and

on each required PORV actuation channel

SR 3.4.12.7 ~~c.~~ Perform~~ance of a~~ CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} ≤ 200 °F

See ITS 3.1.1

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

LCO 3.4.12

Within the limits specified in the COLR, ~~and~~ in addition ~~with the Reactor Coolant System drained below the hot leg centerline,~~ one charging pump shall be ~~rendered inoperable~~.*

capable of injecting into the RCS

a maximum of

M01

APPLICABILITY: MODE 5.

ACTION:

See ITS 3.1.1

If the SHUTDOWN MARGIN requirements cannot be met, immediately initiate and continue boration at ≥ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN requirements of Specification 3.1.1.2 shall be determined:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable.

See ITS 3.1.4

If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

See ITS 3.1.1

SR 3.4.12.2

~~c.~~ ~~At least once per 24 hours~~, ~~when the Reactor Coolant System is drained below the hot leg centerline,~~ ~~by consideration of the factors in 4.1.1.2.b and~~ ~~by~~ verify~~ing at least~~ one charging pump is ~~rendered inoperable~~.*

In accordance with the Surveillance Frequency Control Program

A06

a maximum of

capable of injecting into the RCS

M01

~~* Breaker racked out.~~

LA03

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. In MODES 3* and 4#, one ECCS subsystem composed of one OPERABLE high pressure safety injection pump and one OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal.

See ITS 3.5.3

300

LCO 3.4.12

~~b.~~ Prior to decreasing the reactor coolant system temperature below ~~270~~°F a maximum of only one high pressure safety injection pump shall be OPERABLE with its associated header stop valve open.

M01

LCO 3.4.12 NOTE 3

~~c.~~ Prior to decreasing the reactor coolant system temperature below 236°F all high pressure safety injection pumps shall be disabled and their associated header stop valves closed except as allowed by Specifications 3.1.2.1 and 3.1.2.3.

APPLICABILITY: MODES 3* and 4.

Applicability

MODES 5 and 6 when the Pressurizer manway cover is in place and the reactor vessel head is on.

ACTION:

a. With no ECCS subsystems OPERABLE in MODES 3* and 4#, immediately restore one ECCS subsystem to OPERABLE status or be in COLD SHUTDOWN within 20 hours.

300

ACTION A

b. With RCS temperature below ~~270~~°F and with more than the allowed high pressure safety injection pump OPERABLE or injection valves and header isolation valves open, immediately disable the high pressure safety injection pump(s) or close the header isolation valves.

M01

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

d. LCO 3.0.4.b is not applicable to ECCS High Pressure Safety Injection subsystem when entering MODE 4.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

SR 3.4.12.1

~~4.5.3.2~~ The high pressure safety injection pumps shall be verified inoperable and the associated header stop valves closed prior to decreasing below the above specified Reactor Coolant System temperature and once per month when the Reactor Coolant System is at refueling temperatures.

* With pressurizer pressure < 1750 psia.

\# REACTOR COOLANT SYSTEM cold leg temperature above 250°F.

Low Temperature Overpressure Protection (LTOP) System

A01

Add LCO 3.4.12, HPSI and charging requirements

Add LCO 3.4.12, SIT requirements

M01

M02

Add LCO 3.4.12 Note 3
Power operated relief valves (PORVs) shall only be used when all RCS cold leg temperatures are ≥ 149°F during cooldown and when all RCS cold leg temperatures are ≥ 60°F during heatup.

A05

REACTOR COOLANT SYSTEM

~~OVERPRESSURE PROTECTION SYSTEMS~~

LIMITING CONDITION FOR OPERATION

an RCS vent of ≥

LCO 3.4.12.d ~~3.4.9.3 Unless the~~ RCS ~~is~~ depressurized and ~~vented by at least~~ 3.58 square inches, ~~at least one of the following overpressure protection systems shall be OPERABLE:~~

OPERABLE

s

LCO 3.4.12.a ~~a.~~ Two power-operated relief valves (PORVs) with ~~a~~ lift setting ~~of~~

≤

~~less than or equal to~~ 490 psia ~~and with their associated block valves open. These valves may only be used to satisfy low temperature overpressure protection (LTOP) when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.~~

LA01

LCO 3.4.12.b ~~b.~~ Two shutdown cooling relief valves (SDCRVs) with a lift setting

s

~~of less than or equal to~~ 350 psia~~.~~

≤

≤

LCO 3.4.12.c ~~c.~~ One PORV with ~~a~~ lift setting ~~of less than or equal to~~ 490 psia and ~~with its associated block valve open in conjunction with the use of~~ one SDCRV with ~~a~~ lift setting ~~of less than or equal to~~ 350 psia. ~~This combination may only be used to satisfy LTOP when the~~ RCS cold leg temperature ~~is greater than the temperature listed in Table 3.4-4.~~

LA01

≤

Applicability **APPLICABILITY:** MODES 4#, 5 and 6.

when the reactor vessel head is on

L03

ACTION:

One required

one required

ACTION E ~~a.~~ ~~With either a~~ PORV or ~~an~~ SDCRV ~~being used for LTOP~~ inoperable, restore ~~at least two overpressure protection devices~~ to OPERABLE status within 7 days or:

required PORV or required SDCRV

and establish an RCS vent of ≥

ACTION F ~~1.~~ Depressurize ~~and vent~~ the RCS ~~with a minimum vent area of~~ 3.58 square inches within ~~the next 8~~ hours; OR

12

L01

~~2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3 within the next 8 hours.~~

A02

ACTION F ~~b.~~ ~~With none of the overpressure protection devices being~~ used for LTOP OPERABLE, within the next ~~eight hours~~ either:

12

L01

ACTION F ~~1. Restore at least one overpressure protection device to OPERABLE status or vent the RCS~~; OR

Depressurize the RCS and establish an RCS vent of ≥ 3.58 square inches

~~2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3~~.

A02

Is ≤ 240°F following entry from MODE 3 and when any RCS cold leg temperature is ≤ 252°F following entry from MODE 5.

A05

when any RCS

Applicability # ~~With~~ cold leg temperature ~~within the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3~~.

A01

Low Temperature Overpressure Protection (LTOP) System

REACTOR COOLANT SYSTEM

~~OVERPRESSURE PROTECTION SYSTEMS~~

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

~~c.~~ ~~In the event either the PORVs, SDCRVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, SDCRVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.~~

L02

LCO 3.4.12 ACTIONS Note

~~d.~~ LCO 3.0.4.b is not applicable to PORVs when entering MODE 4.

SURVEILLANCE REQUIREMENTS

~~4.4.9.3.1~~ ~~Each PORV shall be demonstrated OPERABLE by:~~

~~a.~~ ~~In addition to the requirements of the INSERVICE TESTING PROGRAM, operating the PORV through one complete cycle of full travel in accordance with the Surveillance Frequency Control Program.~~

LA02

A01

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

Insert SR 3.4.12.1, SR 3.4.12.2 — M01

Insert SR 3.4.12.3 — M02

SR 3.4.12.6 ~~b.~~ Performance of a CHANNEL FUNCTIONAL TEST on each required ~~the~~ PORV ~~actuation channel~~, ~~but~~ excluding valve ~~operation~~, ~~within 31 days prior to entering a condition in which the PORV is required OPERABLE and~~ in accordance with the Surveillance Frequency Control Program ~~thereafter when the PORV is required OPERABLE~~. A03

SR 3.4.12.7 ~~c.~~ Performance of a CHANNEL CALIBRATION on each required ~~the~~ PORV actuation channel in accordance with the Surveillance Frequency Control Program.

SR 3.4.12.5 ~~d.~~ Verify~~ing the~~ PORV ~~isolation~~ block valve is open for each required PORV in accordance with the Surveillance Frequency Control Program ~~when the PORV is being used for overpressure protection~~.

SR 3.4.12.4 ~~4.4.9.3.2~~ ~~The~~ Verify required RCS vent~~(s) shall be verified to be~~ ≥ 3.58 square inches is open in accordance with the Surveillance Frequency Control Program* ~~when the vent(s) is being used for overpressure protection.~~

SR 3.4.12.4 * Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open in accordance with the Surveillance Frequency Control Program.

A01

~~TABLE 3.4-3~~

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

following entry from MODE 5

following entry from MODE 3

A05

~~Operating Period, EFPY~~	Cold Leg Temperature, °F	
	~~During Heatup~~	~~During Cooldown~~
~~≤ 55~~	≤ 252	≤ 240

LCO 3.4.12 Applicability

A04

~~TABLE 3.4-4~~

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

~~Operating Period EFPY~~	Cold Leg Temperature, °F	
	During Heatup	During Cooldown
~~≤ 55~~	60	149

LCO 3.4.12 Note 3

A04

~~DELETED~~

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODE 5.

See ITS 3.1.1

ACTION:

With the SHUTDOWN MARGIN outside the COLR limits, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

See ITS 3.1.4

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

See ITS 1.1

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

See ITS 3.1.1

SR 3.4.12.2

In accordance with the Surveillance Frequency Control Program

c. ~~At least once per 24 hours~~, ~~when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2b and~~ ~~by~~ verify~~ing~~ ~~at least two~~ charging pump~~s are rendered inoperable~~ ~~by racking out their motor circuit breakers~~.

A06

M01

is capable of injecting into the RCS

a maximum of one

LA03

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Sump Recirculation Actuation Signal.

See ITS 3.5.3

APPLICABILITY: MODES 3* and 4#.

Applicability

Footnote # shall remain applicable in MODES 5 and 6 when the Pressurizer manway cover is in place and the reactor vessel head is on.

ACTION:

a. With no ECCS subsystems OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

c. LCO 3.0.4.b is not applicable to ECCS High Pressure Safety Injection subsystem when entering MODE 4.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

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* With pressurizer pressure less than 1750 psia.

LCO 3.4.12

\# ~~One HPSI shall be rendered inoperable prior to entering MODE 5.~~

M01

DISCUSSION OF CHANGES
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 2 only:** CTS 3.4.9.3 Action a. and b. state, in part, that given the Conditions specified, depressurize and vent the RCS, or be at a temperature above the low temperature RCS overpressure protection range of Table 3.4-3 within the 8 hours. ITS 3.4.12, ACTION F retains the requirement to depressurize the RCS and establish an RCS vent; however, it does not contain the requirement to be at a temperature above the low temperature RCS overpressure protection range of Table 3.4-3.

The purpose of CTS 3.4.9.3 Action a.2 and b.2 is to place the unit in a condition in which the equipment is no longer required. CTS 3.4.9.3 Action a.1 provides appropriate actions (depressurize and vent the RCS) to take with an inoperable PORV or an SDCRV being used for LTOP, and at least one of the two overpressure protection devices is not restored to OPERABLE status within 7 days. CTS 3.4.9.3 Action b.1 provides appropriate actions (depressurize and vent the RCS) to take with none of the overpressure protection devices being used for LTOP are OPERABLE. The overpressure protection devices being used for LTOP are not required to be OPERABLE when the RCS cold leg temperatures are > 252°F, the temperature referenced in CTS Table 3.4-3, because the Mode of Applicability is exited. Therefore, in accordance with CTS 3.0.2 (ITS LCO 3.0.2), when RCS cold leg temperatures are > 252°F, the CTS Actions are no longer required to be completed.

These changes are designated as administrative changes and are acceptable because they maintain the current requirement to place the unit in a condition in which the overpressure protection devices (PORVs and SDCRVs) are no longer required consistent with the requirements of CTS 3.0.2 (ITS LCO 3.0.2) and, therefore, do not result in a technical change to the CTS.

A03 **Unit 2 only:** CTS 4.4.9.3.1.b states, in part, that each PORV shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, “within 31 days prior to entering a condition in which the PORV is required OPERABLE” and in accordance with the Surveillance Frequency Control Program thereafter when the PORV is required OPERABLE. The statement “within 31 days prior to entering a condition in which the PORV is required OPERABLE,” is not included in ITS SR 3.4.16.6. This changes the CTS by removing a Frequency which is duplicative of CTS 4.0.4 (ITS SR 3.0.4). ITS SR 3.0.4 states, in part, that entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency. Therefore,

ITS SR 3.0.4 will continue to require the Surveillance to be performed within 31 days prior to entering a condition in which the PORV is required OPERABLE.

This change is designated as an administrative change and is acceptable because it does not result in technical changes to the CTS.

A04 **Unit 2 only:** CTS 3.4.9.3 references Table 3.4-3, Low Temperature Overpressure Protection (LTOP) Range, and Table 3.4-4, Minimum cold Leg Temperature for PORV Use for LTOP. The "Operating Period, EFPY" and its value of "≤ 55" is deleted from both Tables. ITS 3.4.3, RCS Pressure and Temperature (P/T) Limits, retains the Operating Period and its value in ITS Figures 3.4.3-1 and 3.4.3-2.

This change is designated as an administrative change and is acceptable because it does not result in technical changes to the CTS.

A05 **Unit 2 only:** CTS 3.4.9.3 references Table 3.4-3, Low Temperature Overpressure Protection (LTOP) Range, and Table 3.4-4, Minimum cold Leg Temperature for PORV Use for LTOP. Table 3.4-3 provides RCS cold leg temperature requirements for LTOP protection. CTS Table 3.4-3 is deleted, and the specific values are provided in the ITS 3.4.12 Applicability. To ensure LTOP is applicable when required, "during cooldown" specified in Table 3.4-3 is changed to "following entry from MODE 3" and "during heatup" specified in Table 3.4-3 is changed to "following entry from MODE 5." This change in presentation maintains the intent that during a plant cooldown when transitioning from MODE 3 to MODE 4, LTOP is required when RCS temperature is ≤ 240°F and remains in effect until RCS temperature reaches 252°F during a plant heatup when transitioning from MODE 5 to MODE 4 to MODE 3. CTS Table 3.4-4 provides minimum RCS cold leg temperature requirements for PORV use for LTOP. Table 3.4-4 is deleted and the values in the table are provided as ITS LCO 3.4.12 Note 3.

These presentation changes are designated as an administrative change and are acceptable because they do not result in technical changes to the CTS.

A06 CTS 4.1.1.2.c requires, in part, with the RCS drained below the hot leg centerline, a verification of SDM once per 24 hours by consideration of the factors in 4.1.1.2.b. CTS 4.1.1.2.b also requires a verification of SDM in accordance with the Surveillance Frequency Control Program while in MODE 5 regardless of the drained status of the RCS. The CTS 4.1.1.2.b Frequency specified in the Surveillance Frequency Control Program is 24 hours. ITS 3.4.12 does not include a verification of SDM once per 24 hours with the RCS drained below the hot leg centerline because it is duplicative to the requirement of CTS 4.1.1.2.b which is retained in ITS (SR 3.1.1.1). This change is designated as an administrative change and is acceptable because removal of a duplicative surveillance requirement does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.5.3 limits the OPERABILITY of high pressure safety injection (HPSI) pumps during LTOP conditions based on temperature. Unit 2 CTS 3.5.3, Footnote # requires rendering a HPSI pump inoperable prior to entering MODE 5. Neither Unit 1 CTS 3.4.13 nor Unit 2 CTS 3.4.9.3 (i.e., LTOP Specifications) require a restriction on HPSI pumps. Additionally, Unit 1 CTS 3.4.13 and 3.5.3, and Unit 2 CTS 3.4.9.3 and 3.5.3 do not place any limits on charging pumps. Unit 1 CTS 3.1.1.2 and Footnote * requires, in part, with the RCS drained below the hot leg centerline one charging pump shall be rendered inoperable with the breaker racked out and requires a verification once per 24 hours as specified in CTS 4.1.1.2.c. Unit 2 CTS 4.1.1.2.c provides a similar requirement to verify at least two charging pumps are rendered inoperable when the RCS is drained below the hot leg centerline. ITS LCO 3.4.12 states, in part, that an LTOP System shall be OPERABLE with a maximum of one HPSI pump and one charging pump capable of injecting into the RCS. The ITS LCO contains a Note (Note 1) that states two charging pumps may be made capable of injecting for ≤ 1 hour for pump swap operations. ITS 3.4.12 ACTION A states that if two HPSI pumps are capable of injecting into the RCS, immediately initiate action to verify a maximum of one HPSI pump is capable of injecting into the RCS. ITS 3.4.12 ACTION B states that if two or more charging pumps are capable of injecting into the RCS, immediately initiate action to verify a maximum of one charging pump is capable of injecting into the RCS. If the HPSI pumps or charging pumps Required Action is not met, then ITS 3.4.12 ACTION G states depressurize the RCS and establish an RCS vent of ≥ 1.75 square inches (Unit 1) and ≥ 3.58 square inches (Unit 2). Additionally, ITS SR 3.4.12.1 requires verification that a maximum of one HPSI pump is capable of injecting into the RCS and ITS SR 3.4.12.2 requires verification that a maximum of one charging pump is capable of injecting into the RCS. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per Unit 1 CTS 6.8.4.o and Unit 2 CTS 6.8.4.q. Therefore, SR 3.4.12.1 and SR 3.4.12.2 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program. An initial Frequency of 12 hours will be established for SR 3.4.12.1 consistent with the ISTS SR 3.4.12.1 and an initial Frequency of 24 hours will be established for SR 3.4.12.2 consistent with the Frequency specified in CTS 4.1.1.2.c.

This changes the CTS by requiring all but one charging pump be made incapable of injecting into the RCS during LTOP conditions and provides additional restriction on HPSI pumps during LTOP conditions (i.e., requiring one HPSI pump to be incapable of injecting into the RCS instead of just rendering the pump inoperable) and expands the temperature range on when one HPSI pump must be incapable of injecting into the RCS. This also changes the CTS by adding appropriate actions and Surveillances regarding the HPSI and charging pumps.

The purpose of the LCO is to provide RCS overpressure protection at low temperature conditions by having minimum coolant input capability and adequate RCS pressure relief capacity. Limiting coolant input capability is accomplished by isolating the safety injection tanks (SITs) and restricting high pressure pumps capable of injection into the RCS to a maximum of one high pressure safety injection (HPSI) pump and one charging pump. See DOC M02 for SIT

discussion of change. The Completion Times are reasonable for the ACTIONS to be performed and minimize the time in which the design assumptions for the LTOP System are not being met. The Surveillance Frequency of in accordance with the Surveillance Frequency Control Program considers operating practice to regularly assess potential degradation and to verify operation within the safety analysis.

This change is designated as more restrictive because it adds additional requirements to the CTS.

M02 Unit 1 CTS 3.4.13 and Unit 2 CTS 3.4.9.3 do not place any limits on the safety injection tanks (SITs) during LTOP conditions. ITS LCO 3.4.12 states that the SITs shall be isolated. The ITS LCO contains a Note (Note 2) that states, "SIT may be unisolated when SIT pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in ITS 3.4.3, “RCS Pressure and Temperature (P/T) Limits.” ITS 3.4.12 ACTION C states that if a SIT is not isolated when SIT pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed by the P/T limit curves in LCO 3.4.3, the affected SIT must be isolated within 1 hour. If this isolation is not accomplished, ITS 3.4.12 ACTION D states that the RCS cold leg temperature must be increased to > 300°F (Unit 1) and > 252°F (Unit 2) or the affected SIT must be depressurized and vented within 12 hours. ITS SR 3.4.12.3 requires verification that each SIT is isolated. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per Unit 1 CTS 6.8.4.o and Unit 2 CTS 6.8.4.q. Therefore, SR 3.4.12.3 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 12 hours consistent with the ISTS SR 3.4.12.3. This changes the CTS by adding specific limits on the SITs during LTOP conditions, including appropriate ACTIONS and Surveillance Requirements.

The purpose of the LCO is to provide RCS overpressure protection at low temperature conditions. These changes are necessary because the LTOP analyses assume that the SITs are isolated and therefore not capable of initiating a mass addition transient. The Completion Times are reasonable for the ACTIONS to be performed and minimize the time in which the design assumptions for the LTOP System are not being met. The SR 3.4.12.3 Frequency of in accordance with the Surveillance Frequency Control Program considers operating practice to regularly assess potential degradation and to verify operation within the safety analysis.

The initial frequency established in accordance with the SFCP will be 12 hours for SR 3.1.4.3. See FPL (PSL Unit 1 and Unit 2) “Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program” (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066).

This change is designated as more restrictive because it adds additional requirements to the CTS.

M03 **Unit 1 only:** CTS 3.4.13 Action c. states, in part, that with both PORVs inoperable, actions must be completed within 24 hours. ITS 3.4.12 ACTION G requires the RCS to be depressurized and to establish an RCS vent within 12 hours. Optionally, to exit Condition G, one PORV can be restored to OPERABLE status within 12 hours, under the same conditions. This changes the CTS by allowing 12 hours instead of 24 hours to perform the necessary actions.

The purpose of CTS Actions a. and b. is to place the unit in a condition in which the PORVs are not needed. This change is acceptable because the time with inadequate pressure relief capability is minimized to the time required to establish an adequate RVS vent path. Twelve hours is a sufficient amount of time to allow to plan and execute the maintenance activity of opening an RCS vent. This change allows the necessary activities to be performed in a controlled manner. This change is designated as more restrictive because the required response was changed from 24 hours to 12 hours.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 **Unit 2 only:** (*Type 1 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements*) CTS 3.4.9.3 requires, in part, that the PORVs be OPERABLE with lift settings as specified and their associated block valves open. ITS LCO 3.4.12 does not explicitly require the associated block valve of a required PORV to be open. This changes the CTS by moving the procedural detail that the PORV block valves must be open to the ITS Bases.

The removal of these details for meeting Technical Specification requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for the required PORVs to be OPERABLE. The ITS definition of OPERABLE requires other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). For a PORV to be capable of performing its relief function, its associated block valve must be open to support OPERABILITY. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 **Unit 2 only:** *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 4.4.9.3.1.a requires, in part, that the PORVs be operated through one complete cycle of full travel in accordance with the Surveillance Frequency Control Program. ITS LCO 3.4.12 does not explicitly require the PORVs be manually operated through one complete cycle of full travel. This changes the CTS by moving the Surveillance Requirement that manually operates the PORVs through one complete cycle to the Technical Requirements Manual (TRM).

The removal of this Surveillance Requirement for meeting Technical Specification requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the PORV block valve, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION Surveillances to demonstrate automatic actuation for the PORVs to be OPERABLE consistent with the ISTS. The ITS definition of OPERABLE requires other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). For a PORV to be capable of performing its pressure relief function, its automatic relief actuation must be demonstrated to support OPERABILITY. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a Surveillance Requirement for meeting Technical Specification requirements is being removed from the Technical Specifications. | R1

LA03 (*Type 1 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements*) Unit 1 CTS 3.1.1.2 requires, in part, with the RCS drained below the hot leg centerline one charging pump shall be rendered inoperable. Unit 1 CTS 4.1.1.2.c also requires verification that at least one charging pump is rendered inoperable. The word inoperable is modified by Footnote * that clarifies the breaker racked out. Unit 2 CTS 4.1.1.2.c provides a similar requirement to render at least two charging pumps inoperable by racking out their motor circuit breakers. ITS SR 3.4.12.2 requires verifying a maximum of one charging pump is capable of injecting into the RCS. This changes the CTS by moving the procedural detail of how to make a charging pump inoperable (i.e., incapable of injecting into the RCS) to the ITS Bases.

The removal of these details for meeting Technical Specification requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for limiting the number of charging pumps capable of injecting into the RCS. The charging pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may also be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS.

This may be accomplished through the pump control switch being placed in stop and at least one valve in the discharge flow path being closed. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 – Relaxation of Completion Time)* Unit 1 CTS 3.4.13 Action a. states, in part, that with one PORV inoperable in MODE 4 for > 7 days depressurize and vent the RCS within the next 8 hours. Unit 1 CTS 3.4.13 Action b. states, in part, that with one PORV inoperable in MODES 5 or 6 for > 24 hours depressurize and vent the RCS within a total of 32 hours (essentially 8 hours to depressurize and vent the RCS). Unit 1 ITS 3.4.12 ACTION G requires the RCS to be depressurized and to establish an RCS vent within 12 hours under the same conditions. This changes the CTS by allowing 12 hours instead of 8 hours to perform the required action.

Unit 2 CTS 3.4.9.3 Action a. states, in part, that with either a PORV or an SDCRV being used for LTOP inoperable for > 7 days either depressurize and vent the RCS or be at a temperature above the LTOP enabling temperature of 252°F within the next 8 hours. Unit 2 CTS 3.4.9.3 Action b. states, in part, that with none of the overpressure protection devices being used for LTOP OPERABLE, either restore at least one overpressure protection device to OPERABLE status, or vent the RCS, or be at a temperature above the LTOP enabling temperature of 252°F, within the next 8 hours. Unit 2 ITS 3.4.12 ACTION F requires the RCS to be depressurized and to establish an RCS vent within 12 hours under the same conditions. Optionally, to exit Condition F, RCS cold leg temperature can be raised above the LTOP enabling temperature of 252°F or restore the required relief valves to OPERABLE status. This changes the CTS by allowing 12 hours instead of 8 hours to perform the necessary actions.

The purpose of the Unit 1 CTS Actions a. and b. is to place the unit in a condition in which the PORVs are not needed for LTOP. Similarly, the purpose of the Unit 2 CTS Actions a. and b. is to place the unit in a condition in which the PORVs and SDC relief valves are not needed for LTOP. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition and considers the low probability of a DBA occurring during the allowed Completion Time. Any reduction in the margin of safety due to the extending the time to perform the remedial actions by 4 hours is offset by providing a reasonable time to perform the action to establish an adequate RCS vent path in a controlled manner, thus, minimizing the likelihood of human performance errors that may exacerbate the degraded condition. Twelve hours is a sufficient amount of time to allow to cool and depressurize the RCS (following the unit cooldown rate limits), change MODES, and plan and execute

the maintenance activity of establishing an RCS vent path, or raise RCS cold leg temperature above the LTOP enabling temperature. This change allows the necessary activities to be performed in a controlled manner. This change is designated as less restrictive because additional time is allowed to complete Required Actions than was allowed in the CTS.

L02 *(Category 9 – Deletion of Reporting Requirements)* Unit 1 CTS 3.4.13 Action e and Unit 2 CTS 3.4.9.3 Action c. state that in the event either the PORVs, SDCRVs (Unit 2), or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, SDCRVs (Unit 2), or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence. The ITS does not have a similar requirement. This changes the CTS by eliminating a Special Report.

The purpose of Unit 1 CTS 3.4.13 Action e and Unit 2 CTS 3.4.9.3 Action c is to inform the NRC of challenges to the RCS pressure relief capabilities. This change is acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. The regulatory reporting requirements in 10 CFR 50 are adequate to inform the NRC of challenges to the PORVs, the SDCRVs, or RCS vents, when necessary. Neither the safety analysis assumptions or conditions for continued operation are dependent on the NRC review of the provided information. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L03 **Unit 2 only:** *(Category 2 – Relaxation of Applicability)* CTS 3.4.9.3 Applicability states that the LCO is applicable in MODE 6. ITS 3.4.12 Applicability states that the LCO is applicable in MODE 6 when the reactor vessel head is on. This changes the CTS by only requiring the LCO Applicability in MODE 6 when the reactor vessel head is on.

The purpose of CTS 3.4.9.3 is to ensure there is sufficient low temperature overpressurization protection in all conditions. The definition of MODE 6 included in ITS Table 1.1-1 states that MODE 6 is when one or more reactor vessel head closure bolts are less than fully tensioned. The ITS 3.4.12 Applicability states that the LCO is applicable in MODE 6 when the reactor vessel head is on. This MODE 6 Applicability includes the situation when all reactor vessel head closure bolts are removed, and the vessel head is still on. This change is acceptable since a low temperature overpressurization event is precluded because of the large vent path that is established when the reactor vessel head is removed. This change is designated as less restrictive because it relaxes the Applicability in which the CTS applies.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

3.5.3.b, M01
3.1.1.2

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of one high pressure safety injection (HPSI) pump and one charging pump capable of injecting into the RCS and the safety injection tanks (SITs) isolated, and:

M02

----------NOTES----------

M01

1. ~~[~~Two charging pumps~~]~~ may be made capable of injecting for ≤ 1 hour for pump swap operations. (2)

M02

2. SIT may be unisolated when SIT pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in ~~the PTLR~~. (LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits")

3.5.3 Action C

3. All HPSI pumps shall be incapable of injecting into the RCS when any RCS cold leg temperature is ≤ 236°F except when no charging pumps are available for boration in MODES 5 and 6.

---------- (1)

a. Two OPERABLE power operated relief valves (PORVs) with lift settings ~~within the limits specified in the PTLR~~ or (1)

Applicability

b. The RCS depressurized and an RCS vent of ≥ ~~[~~1.3~~]~~ square inches. (2)

1.75

≤ 530 psia when all RCS cold leg temperatures are > 200°F and
≤ 350 psia when any RCS cold leg temperature is ≤ 200°F

3.4.13 Applicability

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ~~less than or equal to the LTOP enable temperature specified in the PTLR,~~ ≤ 300°F (1)
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

----------NOTE----------

3.4.13 Action f.

LCO 3.0.4.b is not applicable to PORVs when entering MODE 4.

CONDITION	REQUIRED ACTION	COMPLETION TIME
M01 A. Two ~~or more~~ HPSI pumps capable of injecting into the RCS.	A.1 Initiate action to verify a maximum of one HPSI pump capable of injecting into the RCS.	Immediately (1)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M01 B. Two or more charging pumps capable of injecting into the RCS.	B.1 Initiate action to verify a maximum of one charging pump capable of injecting into the RCS.	Immediately
M02 C. A SIT not isolated when SIT pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in ~~the PTLR~~. LCO 3.4.3	C.1 Isolate affected SIT.	1 hour 1
D. Required Action and associated Completion Time of Condition C not met.	D.1 Increase RCS cold leg temperature to > ~~[175]~~°F. 300 OR D.2 Depressurize affected SIT to less than the maximum RCS pressure for existing cold leg temperature allowed in ~~the PTLR~~. LCO 3.4.3	12 hours 2 12 hours 1
3.4.13 Action a. E. One required PORV inoperable in MODE 4.	E.1 Restore required PORV to OPERABLE status.	7 days
3.4.13 Action b. F. One required PORV inoperable in MODE 5 or 6. when the reactor vessel head is on	F.1 Restore required PORV to OPERABLE status.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.13 Action c. G. Two required PORVs inoperable. OR Required Action and associated Completion Time of Condition A, [B], D, E, or F not met. OR LTOP System inoperable for any reason other than Condition A, [B], C, D, E, or F.	G.1 Depressurize RCS and establish RCS vent of ≥ [1.3] square inches. (1.75)	12 hours (2) (2) (2)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
M01 SR 3.4.12.1	Verify a maximum of one HPSI pump is capable of injecting into the RCS.	[12 hours OR In accordance with the Surveillance Frequency Control Program] (2)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
M01 4.1.1.2.c	SR 3.4.12.2	Verify a maximum of one charging pump is capable of injecting into the RCS.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
M02	SR 3.4.12.3	Verify each SIT is isolated.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
3.4.13 Action d.	SR 3.4.12.4	Verify required RCS vent ≥ ~~[~~1.3~~]~~ square inches is open. (1.75)	~~[~~ 12 hours for unlocked open vent valve(s) AND 31 days for other vent path(s) ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~	2 2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.13.a	SR 3.4.12.5	Verify PORV block valve is open for each required PORV.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
M03	SR 3.4.12.6	~~NOTE~~ ~~Not required to be performed until [12] hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR.~~		3
4.4.13.b		Perform CHANNEL FUNCTIONAL TEST on each required PORV, excluding actuation.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.4.13.c	SR 3.4.12.7	Perform CHANNEL CALIBRATION on each required PORV actuation channel.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

3.5.3, M01 LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of one high pressure safety injection (HPSI) pump and one charging pump capable of injecting into the RCS and the safety injection tanks (SITs) isolated, and:

M02

----------NOTES----------

M01 1. ~~[~~Two charging pumps~~]~~ may be made capable of injecting for ≤ 1 hour for pump swap operations. 2

M02 2. SIT may be unisolated when SIT pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in ~~the PTLR~~. 1

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"

---

3. Power operated relief valves (PORVs) shall only be used when all RCS cold leg temperatures are ≥ 149°F during cooldown and when all RCS cold leg temperatures are ≥ 60°F during heatup.

3.4.9.3 Table 3.4-4

a. Two OPERABLE ~~power operated relief valves (~~PORVs~~)~~ with lift settings ~~within the limits specified in the PTLR or~~ 1

3.4.9.3 ~~b~~. The RCS depressurized and an RCS vent of ≥ ~~[1.3]~~ 3.58 square inches. 2

d

3.4.9.3 Applicability APPLICABILITY: MODE 4 when any RCS cold leg temperature is ~~less than or equal to the LTOP enable temperature specified in the PTLR,~~ ≤ 240°F following entry from MODE 3 and when any RCS cold leg temperature is ≤ 252°F following entry from MODE 5, 1

MODE 5,

MODE 6 when the reactor vessel head is on.

3.4.9.3 Table 3.4-3

ACTIONS

----------NOTE----------

3.4.9.3.d Action d. LCO 3.0.4.b is not applicable to PORVs when entering MODE 4.

CONDITION	REQUIRED ACTION	COMPLETION TIME
M01 A. Two ~~or more~~ HPSI pumps capable of injecting into the RCS.	A.1 Initiate action to verify a maximum of one HPSI pump capable of injecting into the RCS.	Immediately 1

3.4.9.3.a ≤ 490 psia;

3.4.9.3.b b. Two shutdown cooling relief valves (SDCRVs) with lift settings ≤ 350 psia; 1

3.4.9.3.c c. One PORV with lift setting ≤ 490 psia and one SDCRV with lift setting ≤ 350 psia; or

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M01 B. Two or more charging pumps capable of injecting into the RCS.	B.1 Initiate action to verify a maximum of one charging pump capable of injecting into the RCS.	Immediately
M02 C. A SIT not isolated when SIT pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in ~~the PTLR~~. LCO 3.4.3	C.1 Isolate affected SIT.	1 hour
D. Required Action and associated Completion Time of Condition C not met.	D.1 Increase RCS cold leg temperature to > ~~[175]~~°F. 252 OR D.2 Depressurize affected SIT to less than the maximum RCS pressure for existing cold leg temperature allowed in ~~the PTLR~~. LCO 3.4.3	12 hours 12 hours
E. One required PORV inoperable ~~in MODE 4.~~	E.1 Restore required ~~PORV~~ relief valve to OPERABLE status.	7 days
~~F. One required PORV inoperable in MODE 5 or 6.~~	~~F.1 Restore required PORV to OPERABLE status.~~	~~24 hours~~

3.4.9.3 Action a.

OR

One required SCDRV inoperable

1 2 1 1 1 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C~~. Two required PORVs inoperable. OR Required Action and associated Completion Time of Condition A, ~~[~~B~~]~~, D, E~~, or F~~ not met. OR LTOP System inoperable for any reason other than Condition A, ~~[~~B~~]~~, C, D, E, ~~or F~~.	~~C~~.1 Depressurize RCS and establish RCS vent of ≥ ~~[1.3]~~ square inches.	12 hours

3.4.9.3 Action a.1, b., b.1

3.4.9.3 Action a., b..

F

or

3.58

OR

One required PORV and one required SCDRV inoperable.

OR

Two required SDCRVs inoperable.

1

2

1

2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of one HPSI pump is capable of injecting into the RCS.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

M01

2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
M01 4.1.1.2.c	SR 3.4.12.2	Verify a maximum of one charging pump is capable of injecting into the RCS.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
M02	SR 3.4.12.3	Verify each SIT is isolated.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.4.9.3.2	SR 3.4.12.4	Verify required RCS vent ≥ ~~[1.3]~~ 3.58 square inches is open.	~~[12 hours for unlocked open vent valve(s)~~ ~~AND~~ ~~31 days for other vent path(s)~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.9.3.1.d	SR 3.4.12.5	Verify PORV block valve is open for each required PORV.	[72 hours OR In accordance with the Surveillance Frequency Control Program]	2
M03	SR 3.4.12.6	~~NOTE~~ ~~Not required to be performed until [12] hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR.~~		3
4.4.9.3.1.b		Perform CHANNEL FUNCTIONAL TEST on each required PORV, excluding actuation.	[31 days OR In accordance with the Surveillance Frequency Control Program]	2
4.4.9.3.1.c	SR 3.4.12.7	Perform CHANNEL CALIBRATION on each required PORV actuation channel.	[[18] months OR In accordance with the Surveillance Frequency Control Program]	2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description. Conditions, Required Actions, and Surveillances are renumbered in the ITS, as applicable.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The Note to ISTS SR 3.4.12.6 is not included in the ITS consistent with current licensing basis. The purpose of the ISTS Note is to allow 12 hours after decreasing Reactor Coolant System (RCS) cold leg temperature to less than or equal to the LTOP enable temperature to perform the test. Per the ISTS Bases, this Note is necessary because the test cannot be performed until the unit is in a condition where the PORV lift setpoint can be reduced to the LTOP setting. At St. Lucie Plant Unit 1 and Unit 2, the PORVs are not required to be OPERABLE for opening in MODE 1, 2, or 3 to support a design basis accident or transient. Therefore, the CHANNEL FUNCTIONAL TEST is currently performed prior to decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature. Testing includes steps performed to minimize the potential for inadvertently opening a PORV and depressurizing the RCS during performance (e.g., closing the PORV block valve and deactivating the PORV from opening). As a result, the CHANNEL FUNCTIONAL TEST can be performed prior to placing the unit in a condition where LTOP is required rendering the ISTS Note unnecessary.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperatures. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3 requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but one high pressure safety injection (HPSI) pump and one charging pump incapable of injection into the RCS and isolating the safety injection tanks (SITs). The pressure relief capacity requires either two OPERABLE redundant power operated relief valves (PORVs) or the RCS depressurized and an RCS vent of sufficient size. One PORV or the RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

BASES

BACKGROUND (continued)

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one ~~[~~HPI or~~]~~ charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions. S 1

The LTOP System for pressure relief consists of two PORVs with reduced lift settings or an RCS vent of sufficient size. Two relief valves are required for redundancy. One PORV has adequate relieving capability to prevent overpressurization for the required coolant input capability.

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The actuation logic monitors RCS pressure and determines when the LTOP overpressure setting is approached. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The LCO presents the PORV setpoints for LTOP. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits of the LCO ensures the P/T limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases coolant, the system pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

BASES

BACKGROUND (continued)

~~For an RCS vent to meet the specified flow capacity, it requires removing a pressurizer safety valve, removing a PORV's internals, and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent valve.~~ The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

~~Safety~~ analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits during shutdown. In MODES 1, 2, and 3, and in MODE 4 with any RCS cold leg temperature greater than the LTOP enable temperature ~~specified in the PTLR~~, the pressurizer safety valves prevent RCS pressure from exceeding the Reference 1 limits. At the LTOP enable temperature ~~specified in the PTLR~~ and below, overpressure prevention falls to the OPERABLE PORVs ~~[~~or to a depressurized RCS and a sufficient sized RCS vent~~]~~. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP System will be re-evaluated to ensure its functional requirements can still be satisfied using the PORV method or the depressurized and vented RCS condition.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

~~a.~~ Inadvertent safety injection ~~or~~

~~b. Charging/letdown flow mismatch~~.

Heat Input Type Transients

~~a. Inadvertent actuation of pressurizer heaters,~~

~~b. Loss of shutdown cooling (SDC), or~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~c.~~ Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators. 1

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all but one HPSI pump, and all but one charging pump incapable of injection and

b. Deactivating the SIT discharge isolation valves in their closed positions.

The Reference 3 analyses demonstrate that either one PORV or the RCS vent can maintain RCS pressure below limits when only one HPSI pump and one charging pump are actuated. Thus, the LCO allows only one HPSI pump and one charging pump ~~OPERABLE~~ during the LTOP MODES. Since neither the PORV nor the RCS vent can handle the pressure transient produced from ~~accumulator~~ injection, when RCS temperature is low, the LCO also requires the SITs isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. 1

capable of injecting into the RCS

SIT

The isolated SITs must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of SIT discharge is over a narrower RCS temperature range ~~([175]°F and below)~~ than that of the LCO ~~(less than or equal to the LTOP enable temperature specified in the PTLR and below)~~. 1

Fracture mechanics analyses established the temperature of LTOP Applicability at less than or equal to the LTOP enable temperature ~~specified in the PTLR~~. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. ~~The vessel materials were assumed to have a neutron irradiation accumulation equal to 21 effective full power years of operation.~~ 1

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 4 and 5), requirements by having a maximum of one HPSI pump and one charging pump ~~OPERABLE and SI actuation enabled for these pumps~~. 1

capable of injecting into the RCS

BASES

APPLICABLE SAFETY ANALYSES (continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limits specified in the ~~PTLR~~. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient of one HPSI pump and one charging pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing setpoints, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensure the Reference 1 limits will be met.

The PORV setpoints will be re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement caused by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV represents the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of [~~1.3~~] square inches is capable of mitigating the limiting allowed LTOP overpressure transient. In that event, this ~~-~~size vent maintains RCS pressure less than the maximum RCS pressure on the P/T limit curve.

The RCS vent size will also be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

This LCO is required to ensure that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires that a maximum of one HPSI pump and one charging pump be capable of injecting into the RCS, and the SITs isolated (when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in ~~the PTLR~~).

LCO 3.4.3

three

1

The LCO is modified by ~~two~~ Notes. Note 1 allows ~~[~~two charging pumps~~]~~ to be made capable of injecting for ≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and Surveillance Requirements associated with the swap. The intent is to minimize the actual time that more than ~~[~~one~~]~~ charging pump is physically capable of injection. Note 2 states that SIT isolation is only required when the SIT pressure is greater than or equal to the RCS pressure for the existing temperature, as allowed by the P/T limit curves provided in ~~the PTLR~~. This Note permits the SIT discharge valve surveillance performed only under these pressure and temperature conditions.

2

2

LCO 3.4.3

1

Note 3 states that all HPSI pumps shall be incapable of injecting into the RCS when any RCS cold leg temperature is ≤ 236°F except when no charging pumps are available for boration in MODES 5 and 6. This Note allows a flow path from the refueling water tank to the RCS via a single HPSI pump if the RCS pressure boundary does not exist, or the RCS pressure boundary integrity exists and no charging pumps are OPERABLE. In the latter case, all charging pumps shall be incapable of injecting into the RCS.

The elements of the LCO that provide overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs or

1

b. The depressurized RCS and an RCS vent.

LCO

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set within the limits specified in the ~~PTLR~~ and testing has proven its ability to open at that setpoint, and motive power is available to the two valves and their control circuits.

1

1.75

An RCS vent is OPERABLE when open with an area ≥ ~~[~~1.3~~]~~ square inches.

2

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

BASES

APPLICABILITY

This LCO is applicable in MODE 4 when the temperature of any RCS cold leg is ~~less than or equal to the LTOP enable temperature specified in the PTLR~~, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the LTOP enable temperature and below. When the reactor vessel head is off, overpressurization cannot occur.

≤ 300°F

1

LCO 3.4.3 provides the operational P/T limits for all MODES.
LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 ~~above the LTOP enable temperature specified in the PTLR~~.

when any RCS cold leg temperature is > 300°F

1

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to inoperable PORVs used for LTOP. There is an increased risk associated with entering MODE 4 from MODE 5 with PORVs used for LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1 and B.1

or two or more charging pumps

1

With two or more HPSI pumps capable of injecting into the RCS, overpressurization is possible.

The immediate Completion Time to initiate actions to restore restricted coolant input capability to the RCS reflects the importance of maintaining overpressure protection of the RCS.

C.1, D.1, and D.2

An unisolated SIT requires isolation within 1 hour. This is only required when the SIT pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed in ~~the PTLR~~.

LCO 3.4.3

1

BASES

ACTIONS (continued)

If isolation is needed and cannot be accomplished within 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed within 12 hours. ~~By increasing the RCS temperature to > [175]°F, a SIT pressure of [600] psig cannot exceed the LTOP limits if the tanks are fully injected. Depressurizing the SIT below the LTOP limit stated in the PTLR also protects against such an event.~~ 1

Raising RCS pressure above the SIT pressure or venting the SIT pressure below the RCS pressure, protects against an LTOP event.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

E.1

In MODE 4 when any RCS cold leg temperature is ≤ ~~[285]~~ 300 °F, with one PORV inoperable, two PORVs must be restored to OPERABLE status within a Completion Time of 7 days. Two valves are required to meet the LCO requirement and to provide low temperature overpressure mitigation while withstanding a single failure of an active component. 2

The Completion Time is based on the facts that only one PORV is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

F.1

The consequences of operational events that will overpressure the RCS are more severe at lower temperature (Ref. 6). Thus, one required PORV inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The 24 hour Completion Time to restore two PORVs OPERABLE in MODE 5 or in MODE 6 when the vessel head is on is a reasonable amount of time to investigate and repair several types of PORV failures without exposure to a lengthy period with only one PORV OPERABLE to protect against overpressure events.

BASES

ACTIONS (continued)

G.1

If two required PORVs are inoperable, or if a Required Action and the associated Completion Time of Condition A, B, D, E, or F are not met, or if the LTOP System is inoperable for any reason other than Condition A through Condition F, the RCS must be depressurized and a vent established within 12 hours. The vent must be sized at least ~~[1.3]~~ 1.75 square inches to ensure the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action protects the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel. 2

The Completion Time of 12 hours to depressurize and vent the RCS is based on the time required to place the plant in this condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, only one HPSI pump and all but ~~[~~one~~]~~ charging pump are verified ~~OPERABLE~~ capable of injecting into the RCS with the other pumps locked out with power removed and the SIT discharge incapable of injecting into the RCS. The ~~[~~HPI~~]~~ pump~~[s]~~ and charging pump~~[s]~~ S are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in ~~[pull to lock]~~ stop and caution tag attached, and at least one valve in the discharge flow path being closed and caution tag attached,. 2 1 2 2

~~[The 12 hour interval considers operating practice to regularly assess potential degradation and to verify operation within the safety analysis.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.4.12.4

1.75

SR 3.4.12.4 requires verifying that the required RCS vent is open ≥ ~~[1.3]~~ square inches is proven OPERABLE by verifying its open condition ~~[~~ either:

2

a. Once every 12 hours for a valve that is unlocked open (valves that are sealed or secured in the open position are considered "locked" in this context) or

b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position, a removed pressurizer safety valve, or open manway).

~~OR~~

2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

The passive vent path arrangement must only be open to be OPERABLE. This Surveillance need only be performed if the vent is being used to satisfy the requirements of this LCO. The Frequencies consider operating experience with mispositioning of unlocked and locked vent valves, respectively.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.5

The PORV block valve must be verified open to provide the flow path for each required PORV to perform its function when actuated. The valve can be remotely verified open in the main control room.

The block valve is a remotely controlled, motor operated valve. The power to the valve motor operator is not required to be removed, and the manual actuator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure event.

~~[The 72 hour Frequency considers operating experience with accidental movement of valves having remote control and position indication capabilities available where easily monitored. These considerations include the administrative controls over main control room access and equipment control.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

SR 3.4.12.6

Performance of a CHANNEL FUNCTIONAL TEST is required to verify and, as necessary, adjust the PORV open setpoints. The CHANNEL FUNCTIONAL TEST will verify on a monthly basis that the PORV lift setpoints are within the LCO limit. A successful test of the required contact(s) of a channel relay may be performed by the verification of the

BASES

SURVEILLANCE REQUIREMENTS (continued)

change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. PORV actuation could depressurize the RCS and is not required. ~~[The 31 day Frequency considers experience with equipment reliability.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

~~A Note has been added indicating this SR is required to be performed [12] hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR. The test cannot be performed until the RCS is in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.~~ 3

SR 3.4.12.7

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required to adjust the whole channel so that it responds and the valve opens within the required LTOP range and with accuracy to known input.

~~[The [18] month Frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ~~FSAR, Section [15]~~. WCAP-17197-NP, "St. Lucie Unit 1 RCS Pressure and Temperature Limits and Low-Temperature Overpressure Protection Report for 54 Effective Full Power Years," Revision 1, January 2021. 1 2
4. 10 CFR 50.46.
5. 10 CFR 50, Appendix K.
6. Generic Letter 90-06.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperatures. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3 requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but one high pressure safety injection (HPSI) pump and one charging pump incapable of injection into the RCS and isolating the safety injection tanks (SITs). The pressure relief capacity requires ~~either~~ two OPERABLE redundant power operated relief valves (PORVs) or the RCS depressurized and an RCS vent of sufficient size. One ~~PORV~~ or the RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

any combination of
and shutdown cooling relief valves (SDCRVs).
relief valve

1

1

BASES

BACKGROUND (continued)

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one ~~[~~HPI or~~]~~ charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

S

relief valves including

The LTOP System for pressure relief consists of two PORVs with reduced lift settings or an RCS vent of sufficient size. Two relief valves are required for redundancy. One ~~PORV~~ has adequate relieving capability to prevent overpressurization for the required coolant input capability.

, two SDCRVs, one PORV with reduced settings and one SDCRV,

relief valve

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The actuation logic monitors RCS pressure and determines when the LTOP overpressure setting is approached. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

1

The LCO presents the PORV setpoints for LTOP. ~~The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient.~~ Having the setpoints of both valves within the limits of the LCO ensures the P/T limits will not be exceeded in any analyzed event.

required

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases coolant, the system pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

SDCRV Requirements

The LCO presents the SDCRV setpoints for LTOP. The SDCRVs may be used to provide for LTOP protection when a SDC loop is aligned to the RCS.

Each SDCRV opens when the RCS pressure exceeds the relief valve setpoint, eliminating the potential for water volume changes and the resulting pressurization effects. The relief valve capacity is greater than the water volume that can be injected into the RCS following a spurious safety injection actuation signal (SIAS) that starts two HPSI pumps, three charging pumps, and isolates the letdown control valves.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

BASES

BACKGROUND (continued)

~~For an RCS vent to meet the specified flow capacity, it requires removing a pressurizer safety valve, removing a PORV's internals, and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent valve.~~ The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

~~Safety~~ analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits during shutdown. In MODES 1, 2, and 3, and in MODE 4 with any RCS cold leg temperature greater than ~~the LTOP enable temperature specified in the PTLR~~, the pressurizer safety valves prevent RCS pressure from exceeding the Reference 1 limits. ~~At the LTOP enable temperature specified in the PTLR and below~~, overpressure prevention falls to the OPERABLE ~~PORVs~~ ~~[~~or to a depressurized RCS and a sufficient sized RCS vent~~]~~. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP System will be re-evaluated to ensure its functional requirements can still be satisfied using the ~~PORV~~ method or the depressurized and vented RCS condition.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

~~a.~~ Inadvertent safety injection ~~or~~

~~b. Charging/letdown flow mismatch~~.

Heat Input Type Transients

~~a. Inadvertent actuation of pressurizer heaters,~~

~~b.~~ ~~Loss of shutdown cooling (SDC)~~, or

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~c.~~ Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators. 1

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all but one HPSI pump, and all but one charging pump incapable of injection and

b. Deactivating the SIT discharge isolation valves in their closed positions.

relief valve

The Reference 3 analyses demonstrate that either one ~~PORV~~ or the RCS vent can maintain RCS pressure below limits when only one HPSI pump and one charging pump are actuated. Thus, the LCO allows only one HPSI pump and one charging pump ~~OPERABLE~~ during the LTOP MODES. Since neither the ~~PORV~~ nor the RCS vent can handle the pressure transient produced from ~~accumulator~~ injection, when RCS temperature is low, the LCO also requires the SITs isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. 1

capable of injecting into the RCS

relief valve

SIT

The isolated SITs must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of SIT discharge is over a narrower RCS temperature range ~~([175]°F and below)~~ than that of the LCO ~~(less than or equal to the LTOP enable temperature specified in the PTLR and below)~~. 1

Fracture mechanics analyses established the temperature of LTOP Applicability at less than or equal to the LTOP enable temperature ~~specified in the PTLR~~. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. ~~The vessel materials were assumed to have a neutron irradiation accumulation equal to 21 effective full power years of operation.~~ 1

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 4 and 5), requirements by having a maximum of one HPSI pump and one charging pump ~~OPERABLE and SI actuation enabled for these pumps~~. 1

capable of injecting into the RCS

BASES

APPLICABLE SAFETY ANALYSES (continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limits specified in the ~~PTLR~~. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient of one HPSI pump and one charging pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing setpoints, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensure the Reference 1 limits will be met.

The PORV setpoints will be re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement caused by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV represents the worst case, single active failure.

SDCRV Performance

The SDCRVs are sized for a flow capacity of 2300 gpm at a lift set pressure of 350 psia. Each of two SDC relief valves is sufficient to provide LTOP during low temperature operation when the SDCRVs are aligned to the RCS.

The SDCRVs are passive components of the shutdown cooling system.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of [~~1.3~~] square inches is capable of mitigating the limiting allowed LTOP overpressure transient. In that event, this ~~-~~size vent maintains RCS pressure less than the maximum RCS pressure on the P/T limit curve.

The RCS vent size will also be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

This LCO is required to ensure that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires that a maximum of one HPSI pump and one charging pump be capable of injecting into the RCS, and the SITs isolated (when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in ~~the PTLR~~).

LCO 3.4.3

three

The LCO is modified by ~~two~~ Notes. Note 1 allows ~~[~~two charging pumps~~]~~ to be made capable of injecting for ≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and Surveillance Requirements associated with the swap. The intent is to minimize the actual time that more than ~~[~~one~~]~~ charging pump is physically capable of injection. Note 2 states that SIT isolation is only required when the SIT pressure is greater than or equal to the RCS pressure for the existing temperature, as allowed by the P/T limit curves provided in ~~the PTLR~~. This Note permits the SIT discharge valve surveillance performed only under these pressure and temperature conditions.

LCO 3.4.3

Note 3 allows the PORVs to be used when all RCS cold leg temperatures are ≥ 149°F during cooldown and when all RCS cold leg temperatures are ≥ 60°F during heatup. As a result, LCO 3.4.12.a and c pressure relief options are not allowed below 149°F during cooldown following a reactor shutdown. The SDCRVs or an RCS vent path of > 3.58 square inches are utilized for LTOP protection below 149°F. This restriction is necessary because the required relief capability of a PORV during a cooldown is not sufficient to overcome coolant input capability. During vessel heatup operations, the relief capability of the PORVs are sufficient. Use of PORVs below 60°F was not analyzed and therefore not acceptable for use below that value.

The elements of the LCO that provide overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs or

two OPERABLE SDCRVs, or one OPERABLE PORV and one OPERABLE SDCRV, and

b. The depressurized RCS and an RCS vent.

LCO

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set within the limits specified in the ~~PTLR~~ and testing has proven its ability to open at that setpoint, and motive power is available to the ~~two~~ valve~~s~~ and their control circuits.

3.58

An RCS vent is OPERABLE when open with an area ≥ ~~[1.3]~~ square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

A SDCRV is OPERABLE for LTOP when its lift setpoint is set within the limits specified in the LCO and the SDC system is aligned to the RCS.

The LTOP enable temperature is dependent on RCS thermal transient conditions. LTOP analyses indicate that LTOP may be enabled at a lower RCS temperature during an RCS cooldown following a reactor shutdown than during an RCS heatup from cold conditions due to the lower reactor vessel fracture toughness when the reactor vessel is cold. Therefore, the LCO is applicable in MODE 4 when any RCS cold leg temperature is less or equal to the LTOP cooldown temperature of 240°F following entry into MODE 4 from MODE 3 and in MODE 4 when any RCS cold leg temperature is less or equal to the LTOP heatup temperature of 252°F following entry into MODE 4 from MODE 5.

BASES

APPLICABILITY

≤ 240°F following entry from MODE 3 and when any RCS cold leg temperature is ≤ 252°F following entry from MODE 5.

This LCO is applicable in MODE 4 when the temperature of any RCS cold leg is ~~less than or equal to the LTOP enable temperature specified in the PTLR~~, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the LTOP enable temperature and below. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES.
LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves ~~that provide overpressure protection~~ during MODES 1, 2, and 3, and MODE 4 ~~above the LTOP enable temperature specified in the PTLR~~.

when all RCS cold leg temperatures are > 230°F to ensure uninterrupted overpressure protection.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to inoperable PORVs used for LTOP. There is an increased risk associated with entering MODE 4 from MODE 5 with PORVs used for LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1 and B.1

or two or more charging pumps

With two or more HPSI pumps capable of injecting into the RCS, overpressurization is possible.

The immediate Completion Time to initiate actions to restore restricted coolant input capability to the RCS reflects the importance of maintaining overpressure protection of the RCS.

C.1, D.1, and D.2

An unisolated SIT requires isolation within 1 hour. This is only required when the SIT pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed in ~~the PTLR~~.

LCO 3.4.3

BASES

ACTIONS (continued)

If isolation is needed and cannot be accomplished within 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed within 12 hours. ~~By increasing the RCS temperature to > [175]°F, a SIT pressure of [600] psig cannot exceed the LTOP limits if the tanks are fully injected. Depressurizing the SIT below the LTOP limit stated in the PTLR also protects against such an event.~~ 1

Raising RCS pressure above the SIT pressure or venting the SIT pressure below the RCS pressure, protects against an LTOP event.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

E.1

or one required SDCRV inoperable,

required relief valves

required

~~In MODE 4 when any RCS cold leg temperature is ≤ [285]°F,~~ with one PORV inoperable, two ~~PORVs~~ must be restored to OPERABLE status within a Completion Time of 7 days. Two valves are required to meet the LCO requirement and to provide low temperature overpressure mitigation while withstanding a single failure of an active component. 1 2

relief valve

The Completion Time is based on the facts that only one ~~PORV~~ is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low. 1

~~F.1~~

~~The consequences of operational events that will overpressure the RCS are more severe at lower temperature (Ref. 6). Thus, one required PORV inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.~~ 1

~~The 24 hour Completion Time to restore two PORVs OPERABLE in MODE 5 or in MODE 6 when the vessel head is on is a reasonable amount of time to investigate and repair several types of PORV failures without exposure to a lengthy period with only one PORV OPERABLE to protect against overpressure events.~~

BASES

ACTIONS (continued)

F

~~G~~.1

or one required PORV and one required SDCRV are inoperable, or two required SDCRVs are inoperable,

or

If two required PORVs are inoperable, or if a Required Action and the associated Completion Time of Condition A, B, D, E, ~~or F~~ are not met, or if the LTOP System is inoperable for any reason other than Condition A through Condition ~~F~~, the RCS must be depressurized and a vent established within 12 hours. The vent must be sized at least ~~[1.3]~~ square inches to ensure the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action protects the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

1

E

3.58

2

The Completion Time of 12 hours to depressurize and vent the RCS is based on the time required to place the plant in this condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, only one HPSI pump and all but ~~[~~one~~]~~ charging pump are verified ~~OPERABLE~~ with the other pumps locked out with power removed and the SIT discharge incapable of injecting into the RCS. The ~~[~~HPI~~]~~ pump~~[s]~~ and charging pump~~[s]~~ are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in ~~[pull to lock]~~ and at least one valve in the discharge flow path being closed.

2

1

capable of injecting into the RCS

S

2

2

stop and control tag attached,

and control tag attached,

~~[The 12 hour interval considers operating practice to regularly assess potential degradation and to verify operation within the safety analysis.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.4.12.4

3.58

SR 3.4.12.4 requires verifying that the required RCS vent is open ≥ ~~[1.3]~~ square inches is proven OPERABLE by verifying its open condition [either:

2

~~a. Once every 12 hours for a valve that is unlocked open (valves that are sealed or secured in the open position are considered "locked" in this context) or~~

~~b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position, a removed pressurizer safety valve, or open manway).~~

~~OR~~

2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

The passive vent path arrangement must only be open to be OPERABLE. This Surveillance need only be performed if the vent is being used to satisfy the requirements of this LCO. The Frequencies consider operating experience with mispositioning of unlocked and locked vent valves, respectively.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.5

The PORV block valve must be verified open to provide the flow path for each required PORV to perform its function when actuated. The valve can be remotely verified open in the main control room.

The block valve is a remotely controlled, motor operated valve. The power to the valve motor operator is not required to be removed, and the manual actuator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure event.

~~[The 72 hour Frequency considers operating experience with accidental movement of valves having remote control and position indication capabilities available where easily monitored. These considerations include the administrative controls over main control room access and equipment control.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

SR 3.4.12.6

Performance of a CHANNEL FUNCTIONAL TEST is required to verify and, as necessary, adjust the PORV open setpoints. The CHANNEL FUNCTIONAL TEST will verify on a monthly basis that the PORV lift setpoints are within the LCO limit. A successful test of the required contact(s) of a channel relay may be performed by the verification of the

BASES

SURVEILLANCE REQUIREMENTS (continued)

change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. PORV actuation could depressurize the RCS and is not required. ~~[The 31 day Frequency considers experience with equipment reliability.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

~~A Note has been added indicating this SR is required to be performed [12] hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR. The test cannot be performed until the RCS is in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.~~ 3

SR 3.4.12.7

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required to adjust the whole channel so that it responds and the valve opens within the required LTOP range and with accuracy to known input.

~~[The [18] month Frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ~~FSAR, Section [15]~~.
4. 10 CFR 50.46.
5. 10 CFR 50, Appendix K.
6. Generic Letter 90-06.

WCAP-16817-NP, "St. Lucie Unit 2 RCS Pressure and Temperature Limits and Low Temperature Overpressure Protection Report for 55 Effective Full Power Years," Revision 2, October 2007. 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.12, BASES, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes have been made to be consistent with changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.12, LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 13

3.4.13, RCS Operational Leakage

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTOR COOLANT SYSTEM

RCS Operational

~~REACTOR COOLANT SYSTEM~~ LEAKAGE

LIMITING CONDITION FOR OPERATION

RCS

LCO 3.4.13 ~~3.4.6.2~~ ~~Reactor Coolant System~~ operational leakage shall be limited to:

LCO 3.4.13.a a. No PRESSURE BOUNDARY LEAKAGE,

LCO 3.4.13.b b. 1 GPM UNIDENTIFIED LEAKAGE,

LCO 3.4.13.c c. 150 gallons per day primary-to-secondary leakage through any one steam generator (SG),

LCO 3.4.13.d d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4.6-1 for each Reactor Coolant System Pressure Isolation Valve identified in Table 3.4.6-1.

See ITS 3.4.14

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTION:

Insert proposed ITS 3.4.13 ACTION A

L01

ACTION C ~~a.~~ ~~With any~~ PRESSURE BOUNDARY LEAKAGE~~, or with~~ primary-to-secondary leakage not within limit, be in ~~at least~~ ~~HOT STANDBY~~ within 6 hours and ~~in~~ ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

MODE 5

MODE 3

RCS

36

not within

ACTION B ~~b.~~ ~~With any Reactor Coolant System~~ operational leakage ~~greater than any one of the above~~ limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE, ~~and Reactor Coolant System Pressure Isolation Valve leakage~~, reduce ~~the~~ leakage ~~rate~~ to within limits within 4 hours or be in ~~at least~~ ~~HOT STANDBY~~ within 6 hours and ~~in~~ ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

ACTION C

MODE 3

MODE 5

36

NOTE

Enter applicable ACTIONS for systems made inoperable by an inoperable pressure isolation valve.

See ITS 3.4.14

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit in 3.4.6.2.e above reactor operation may continue provided that at least two valves, including check valves, in each high pressure line having a non-functional valve are in and remain in the mode corresponding to the isolated condition. Motor operated valves shall be placed in the closed position, and power supplies deenergized. Otherwise, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

~~4.4.6.2~~ ~~Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:~~

~~a.~~ ~~Monitoring the containment atmosphere gaseous and particulate radioactivity in accordance with the Surveillance Frequency Control Program.~~

L02

A01

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

~~b. Monitoring the containment sump inventory and discharge in accordance with the Surveillance Frequency Control Program,~~ L02

Verify RCS operational LEAKAGE is within limits by

SR 3.4.13.1

~~c.~~ *Performance of a ~~Reactor Coolant System~~ water inventory balance in accordance with the Surveillance Frequency Control Program ~~except when operating in the shutdown cooling mode,~~ L03

RCS

~~d.~~ ~~Monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program, and~~ L02

e. Verifying each Reactor Coolant System Pressure Isolation Valve leakage (Table 3.4.6-1) to be within limits:

See ITS 3.4.14

1. Prior to entering MODE 2 after refueling,

2. Prior to entering MODE 2, whenever the plant has been in COLD SHUTDOWN for 7 days or more if leakage testing has not been performed in the previous 9 months,

3. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

4. The provision of Specification 4.0.4 is not applicable for entry into MODE 3 or 4.

f. Whenever integrity of a pressure isolation valve listed in Table 3.4.6-1 cannot be demonstrated the integrity of the remaining check valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in each high pressure line having a leaking valve shall be recorded daily; and

Verify

SR 3.4.13.2

~~g.~~ Primary-to-secondary leakage ~~shall be verified~~ ≤150 gallons per day through any one ~~steam generator~~ in accordance with the Surveillance Frequency Control Program.** SG

SR 3.4.13.1 Note 1* Not required to be performed until 12 hours after establishment of steady state operation.

SR 3.4.13.1 Note 2 Not applicable to primary-to-secondary leakage.

SR 3.4.13.2 Note ** Not required to be performed until 12 hours after establishment of steady state operation.

TABLE 3.4 6-1

See ITS 3.4.14

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

Check Valve No.
V3227
V3123
V3217
V3113
V3237
V3133
V3247
V3143
V3124
V3114
V3134
V3144

NOTES

(a) Maximum Allowable Leakage (each valve):

1. Leakage rates less than or equal to 1.0 gpm are acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount the reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are unacceptable.

(b) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(c) Minimum test differential pressure shall not be less than 150 psid.

REACTOR COOLANT SYSTEM

RCS

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

RCS

LCO 3.4.13 ~~3.4.6.2~~ ~~Reactor Coolant System~~ operational leakage shall be limited to:

LCO 3.4.13.a a. No PRESSURE BOUNDARY LEAKAGE,

LCO 3.4.13.b b. 1 gpm UNIDENTIFIED LEAKAGE,

LCO 3.4.13.c c. 150 gallons per day primary-to-secondary leakage through any one steam generator (SG),

LCO 3.4.13.d d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage (except as noted in Table 3.4-1) at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

See ITS 3.4.14

Applicability **APPLICABILITY:** MODES 1, 2, 3, and 4.

ACTION:

Insert proposed ITS 3.4.13 ACTION A

L01

ACTION C ~~a.~~ ~~With any~~ PRESSURE BOUNDARY LEAKAGE ~~or with~~ primary-to-secondary leakage not within limit, be in ~~at least HOT STANDBY~~ within 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

MODE 5 MODE 3 RCS 36

ACTION B b. ~~With any Reactor Coolant System~~ operational leakage ~~greater than any one of the~~ limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE, ~~and leakage from Reactor Coolant System Pressure Isolation Valves~~, reduce ~~the~~ leakage ~~rate~~ to within limits within 4 hours or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

not within

ACTION C MODE 3 MODE 5 36

NOTE

Enter applicable ACTIONS for systems made inoperable by an inoperable pressure isolation valve.

See ITS 3.4.14

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~d. With RCS leakage alarmed and confirmed in a flow path with no flow indication, commence an RCS water inventory balance within 1 hour to determine the leak rate.~~

L04

SURVEILLANCE REQUIREMENTS

~~4.4.6.2.1 Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:~~

~~a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor in accordance with the Surveillance Frequency Control Program.~~

L02

~~b. Monitoring the containment sump inventory and discharge in accordance with the Surveillance Frequency Control Program.~~

L02

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

Verify RCS operational LEAKAGE is within limits by

SR 3.4.13.1

~~c.~~ *Performance of a ~~Reactor Coolant System~~ RCS water inventory balance in accordance with the Surveillance Frequency Control Program.

~~d. Monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program.~~ L02

SR 3.4.13.2

~~e.~~ Verify~~ing~~ primary-to-secondary leakage is ≤ 150 gallons per day through any one ~~steam generator~~ SG in accordance with the Surveillance Frequency Control Program.**

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve check valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit: See ITS 3.4.14

a. In accordance with the Surveillance Frequency Control Program,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d. Following valve actuation due to automatic or manual action or flow through the valve:

1. Within 24 hours by verifying valve closure, and

2. Within 31 days by verifying leakage rate.

4.4.6.2.3 Each Reactor Coolant System Pressure Isolation Valve motor-operated valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit;

a. In accordance with the Surveillance Frequency Control Program, and

b. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

SR 3.4.13.1 Note 1* Not required to be performed until 12 hours after establishment of steady state operation.

SR 3.4.13.1 Note 2 Not applicable to primary-to-secondary leakage.

SR 3.4.13.2 Note ** Not required to be performed until 12 hours after establishment of steady state operation.

A01

See ITS 3.4.14

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

Check Valve No.		Motor Operated Valve No.
V3217	V3525	V3480
V3227	V3524	V3481
V3237	V3527	V3652
V3247	V3526	V3651
V3259		
V3258		
V3260		
V3261		
V3215		
V3225		
V3235		
V3245		

NOTES

(a) Maximum Allowable Leakage (each valve):

1. Except as noted below leakage rates greater than 1.0 gpm are unacceptable.

2. For motor-operated valves (MOVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount the reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. For motor-operated valves (MOVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are unacceptable.

(b) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(c) Minimum test differential pressure shall not be less than 200 psid.

DISCUSSION OF CHANGES
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.4.6.2 Action a. states, in part, with any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY (i.e., MODE 3) within 6 hours and in COLD SHUTDOWN (i.e., MODE 5) within the following 30 hours. ITS 3.4.13 ACTION A states that if pressure boundary LEAKAGE exists, isolate the affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve within 4 hours. Additionally, ITS 3.4.13 ACTION C states that with the Required Action and associated Completion Time not met, be in MODE 3 within 6 hours and MODE 5 within 36 hours. This changes the CTS by allowing 4 hours to isolate the leakage before requiring Actions to exit the applicable MODES, consistent with Technical Specification Task Force (TSTF) traveler TSTF-554-A, Revision 1, “Revise Reactor Coolant Leakage Requirements.”

TSTF traveler TSTF-554-A incorporated changes to the Standard Technical Specifications (STSs) under the consolidated line item improvement process (CLIIP). TSTF-554-A was approved for use by the NRC as documented in staff Safety Evaluation (SE) dated December 18, 2020 (ADAMS Accession No. ML20322A361, ML20322A024). PSL has reviewed the NRC staff SE and concluded that the justification presented in TSTF-554-A and the SE prepared by the NRC staff are applicable to PSL and justify this change.

TSTF-554-A revised the technical specifications related to reactor coolant system (RCS) operational leakage by adding Condition A that applies when pressure boundary LEAKAGE exists and Required Action A.1 that requires isolation of the affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve, within a Completion Time of 4 hours. If the 4-hour Completion Time cannot be met, the plant must initiate shutdown in accordance with ACTION C. The Required Action requires the flaw to be isolated from the reactor coolant pressure source to prevent further degradation of the flaw, which could result in additional leakage. If Required Action A.1 cannot be completed within the 4-hour Completion Time, ISTS 3.4.13 Condition C requires that the reactor be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences (i.e., be in MODE 3 within 6 hours and MODE 5 within 36 hours. ISTS 3.4.13 Condition A, including its associated Required Action A.1 and Completion Time, is acceptable because it continues to meet the requirements of 10 CFR 50.36(c)(2)(i), by providing remedial actions and shutting down the reactor if the remedial actions cannot be met.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 (Category 5 – Deletion of Surveillance Requirement) Unit 1 CTS 4.4.6.2.a and Unit 2 CTS 4.4.6.2.1.a require monitoring of the containment atmosphere gaseous and particulate radioactivity in accordance with the Surveillance Frequency Control Program. Unit 1 CTS 4.4.6.2.b and Unit 2 CTS 4.4.6.2.1.b require monitoring the containment sump inventory and discharge in accordance with the Surveillance Frequency Control Program. Unit 1 CTS 4.4.6.2.d and Unit 2 CTS 4.4.6.2.1.d require monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program. The ITS does not contain these Surveillance Requirements. This changes the CTS by deleting these Surveillance Requirements.

This change is acceptable because the deleted Surveillance Requirements are not necessary to verify that the LCO is being met. Appropriate Surveillance Requirements continue to be performed in a manner and at a Frequency necessary to give confidence that the LCO is being met. The indications in the deleted Surveillance Requirements are not necessarily indications of failure to meet the LCO on RCS operational leakage. These items do provide useful information and the containment atmosphere gaseous and particulate radioactivity monitors, and the containment sump monitors are required to be OPERABLE by ITS 3.4.15, "RCS Leakage Detection Instrumentation." However, under ITS SR 3.0.1, failure to meet the Surveillance results in failure to meet the LCO. As these indications do not necessarily indicate a failure to meet the LCO, it is not appropriate to retain these indications in this Specification. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L03 **Unit 1 only:** *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.4.6.2.c requires performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program "except when operating in the shutdown cooling mode" and is modified by a footnote that states not required to be performed until 12 hours after establishment of steady state operation. ITS SR 3.4.13.1 does not contain the exception that states "except when operating in the shutdown cooling mode." This changes the CTS by deleting the exception for performance of the Surveillance when operating in the shutdown cooling mode.

The purpose of CTS 4.4.6.2.c is to verify RCS LEAKAGE within the LCO limits to ensure the integrity of the RCPB is maintained. Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, and makeup and letdown. An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the values used to meet the LCO are consistent with the safety analysis. ITS SR 3.4.13.1 retains the Surveillance Requirement to verify RCS operation LEAKAGE is within limits by performance of RCS water inventory balance. This change is designated as less restrictive because a Surveillance required in the CTS will not be required in the ITS.

L04 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* CTS 3.4.6.2 Action d. states that with RCS leakage alarmed and confirmed in a flow path with no flow indication, commence an RCS water inventory balance within 1 hour to determine the leak rate. ITS 3.4.13 does not contain a requirement for an RCS water inventory balance in response to system alarms. This changes the CTS by deleting the explicit requirement for an RCS water inventory balance in response to system alarms.

The purpose of CTS 3.4.6.2 is to verify RCS LEAKAGE within the LCO limits and take appropriate actions in response to operational LEAKGE not within limits, to ensure the integrity of the RCPB is maintained. An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation." ITS 3.4.15 provides ACTIONS that include performance of SR 3.4.13.1 (i.e., RCS water inventory balance). This change is acceptable because the deleted requirement is not necessary to verify that the values used to meet the LCO are consistent with the safety analysis. ITS SR 3.4.13.1 retains the Surveillance Requirement to verify RCS operation LEAKAGE is within limits by performance of RCS water inventory balance. This change is designated as less restrictive because an Action required in the CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

3.4.6.2 LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

3.4.6.2.a a. No pressure boundary LEAKAGE,

3.4.6.2.b b. 1 gpm unidentified LEAKAGE,

3.4.6.2.d c. 10 gpm identified LEAKAGE, and

3.4.6.2.c d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 A. Pressure boundary LEAKAGE exists.	A.1 Isolate affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve.	4 hours
3.4.6.2 Action b. B. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	B.1 Reduce LEAKAGE to within limits.	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.6.2 Action b. C. Required Action and associated Completion Time not met. OR 3.4.6.2 Action a. Primary to secondary LEAKAGE not within limit.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.4.6.2.c SR 3.4.13.1	----------NOTES---------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ----------	
	Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
4.4.6.2.g SR 3.4.13.2	----------NOTE---------- Not required to be performed until 12 hours after establishment of steady state operation. ----------	
	Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

3.4.6.2 LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

3.4.6.2.a a. No pressure boundary LEAKAGE,

3.4.6.2.b b. 1 gpm unidentified LEAKAGE,

3.4.6.2.d c. 10 gpm identified LEAKAGE, and

3.4.6.2.c d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 A. Pressure boundary LEAKAGE exists.	A.1 Isolate affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve.	4 hours
3.4.6.2 Action b. B. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	B.1 Reduce LEAKAGE to within limits.	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.6.2 Action b. C. Required Action and associated Completion Time not met. OR 3.4.6.2 Action a. Primary to secondary LEAKAGE not within limit.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.4.6.2.1.c SR 3.4.13.1	----------NOTES---------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ----------	
	Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.4.6.2.1.e	SR 3.4.13.2	----------NOTE---------- Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	~~[~~ 72 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS LEAKAGE detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from ~~all~~ steam generators (SGs) is ~~[1~~ gallon per minute~~]~~ or increases to ~~[1~~ gallon per minute~~]~~ as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis. (0.25, 0.25, 2)

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The ~~[1~~ gpm~~]~~ primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential. (0.25, 2)

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes ~~the entire~~ ~~[1~~ gpm~~]~~ primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50 or the staff approved licensing basis (i.e., a small fraction of these limits). (0.25, 2)

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

Pressure boundary LEAKAGE is prohibited as the leak itself could cause further RCPB deterioration, resulting in higher LEAKAGE.

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Separating the sources of leakage (i.e., leakage from an identified source versus leakage from an unidentified source) is necessary for prompt identification of potentially adverse conditions, assessment of the safety significance, and corrective action.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE).

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

ACTIONS

A.1

If pressure boundary LEAKAGE exists, the affected component, pipe, or vessel must be isolated from the RCS by a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve within 4 hours. While in this condition, structural integrity of the system should be considered because the structural integrity of the part of the system within the isolation boundary must be maintained under all licensing basis conditions, including consideration of the potential for further degradation of the isolated location. Normal LEAKAGE past the isolation device is acceptable as it will limit RCS LEAKAGE and is included in identified or unidentified LEAKAGE. This action is necessary to prevent further deterioration of the RCPB.

B.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

C.1 and C.2

If primary to secondary LEAKAGE is not within limit, or if any of the Required Actions and associated Completion Times cannot be met, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, ~~[and RCP seal injection and return flows]~~). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. 2

Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown~~, and RCP seal injection and return flows~~. 2

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

~~[The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.~~18~~, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

17

1

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown~~, and RCP seal injection and return flows~~.

1

~~[The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.

2. Regulatory Guide 1.45, May 1973.

3. FSAR, Section ~~[15]~~ 15.4.4. U 2

4. NEI 97-06, "Steam Generator Program Guidelines."

5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS LEAKAGE detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from ~~all~~ steam generators (SGs) is ~~[1~~ gallon per minute~~]~~ or increases to ~~[1~~ gallon per minute~~]~~ as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis. 0.25 0.25 2

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The ~~[1~~ gpm~~]~~ primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential. 0.25 2

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes ~~the entire~~ ~~[1~~ gpm~~]~~ primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50 or the staff approved licensing basis (i.e., a small fraction of these limits). 0.25 2

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

Pressure boundary LEAKAGE is prohibited as the leak itself could cause further RCPB deterioration, resulting in higher LEAKAGE.

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Separating the sources of leakage (i.e., leakage from an identified source versus leakage from an unidentified source) is necessary for prompt identification of potentially adverse conditions, assessment of the safety significance, and corrective action.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE).

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

ACTIONS

A.1

If pressure boundary LEAKAGE exists, the affected component, pipe, or vessel must be isolated from the RCS by a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve within 4 hours. While in this condition, structural integrity of the system should be considered because the structural integrity of the part of the system within the isolation boundary must be maintained under all licensing basis conditions, including consideration of the potential for further degradation of the isolated location. Normal LEAKAGE past the isolation device is acceptable as it will limit RCS LEAKAGE and is included in identified or unidentified LEAKAGE. This action is necessary to prevent further deterioration of the RCPB.

B.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

C.1 and C.2

If primary to secondary LEAKAGE is not within limit, or if any of the Required Actions and associated Completion Times cannot be met, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, ~~[and RCP seal injection and return flows]~~). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. 2

Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown~~, and RCP seal injection and return flows~~. 2

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

~~[The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.~~18~~, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

17

1

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown~~, and RCP seal injection and return flows~~.

1

~~[The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.

2. Regulatory Guide 1.45, May 1973.

3. FSAR, Section ~~[15]~~ 15.4.4. U 2

4. NEI 97-06, "Steam Generator Program Guidelines."

5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.13, BASES, RCS OPERATIONAL LEAKAGE

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.13, RCS OPERATIONAL LEAKAGE

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 14

3.4.14, RCS Pressure Isolation Valve (PIV) Leakage

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTOR COOLANT SYSTEM

RCS PIV

~~REACTOR COOLANT SYSTEM~~ LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator (SG),

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

See ITS 3.4.13

LCO 3.4.14
SR 3.4.14.1

e. Leakage as specified in ~~Table 3.4.6-1~~ for each Reactor Coolant System Pressure Isolation Valve ~~identified in Table 3.4.6-1~~.

APPLICABILITY: MODES 1, 2, 3 and 4.

Add proposed Applicability – MODE 4 exception

L02

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.4.13

b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE, and Reactor Coolant System Pressure Isolation Valve leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE

Add proposed LCO 3.4.14 ACTIONS Note 1

A02

LCO 3.4.14
ACTIONS Note 2

Enter applicable ~~ACTIONS~~ for systems made inoperable by an inoperable pressure isolation valve.

Conditions and Required Actions

Add proposed LCO 3.4.14 Required Action A.1 and A.2 Note

M01

ACTION A

~~c.~~ With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit in 3.4.6.2.e above reactor operation may continue provided that at least two valves, including check valves, in each high pressure line having a non-functional valve are in and remain in the mode corresponding to the isolated condition. Motor operated valves shall be placed in the closed position, and power supplies deenergized. Otherwise, reduce the leakage rate to within limits within ~~4~~ hours or be in ~~at least HOT STANDBY~~ within 6 hours and ~~in COLD SHUTDOWN~~ within ~~the following 30~~ 72 hours. 36

Required Action A.2

ACTION B

MODE 3

MODE 5

L01

SURVEILLANCE REQUIREMENTS

See ITS 3.4.13

4.4.6.2 Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity in accordance with the Surveillance Frequency Control Program.

A01

REACTOR COOLANT SYSTEM

RCS PIV

~~REACTOR COOLANT SYSTEM~~ LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the containment sump inventory and discharge in accordance with the Surveillance Frequency Control Program,

See ITS 3.4.13

c. *Performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program except when operating in the shutdown cooling mode,

d. Monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program, and

SR 3.4.14.1

~~e.~~ Verifying each Reactor Coolant System Pressure Isolation Valve leakage ~~(Table 3.4.6-1)~~ to be within limits:

LA01

1. Prior to entering MODE 2 after refueling,

2. Prior to entering MODE 2, whenever the plant has been in COLD SHUTDOWN for 7 days or more if leakage testing has not been performed in the previous 9 months,

Add proposed SR 3.4.14.1 Frequency – in accordance with the Surveillance Frequency Control Program, and within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

M02

~~3. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.~~

L03

SR 3.4.14.1 Note 1

~~4.~~ The provision of Specification 4.0.4 is not applicable for entry into MODE 3 or 4.

~~f. Whenever integrity of a pressure isolation valve listed in Table 3.4.6-1 cannot be demonstrated the integrity of the remaining check valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in each high pressure line having a leaking valve shall be recorded daily; and~~

L04

g. Primary-to-secondary leakage shall be verified ≤150 gallons per day through any one steam generator in accordance with the Surveillance Frequency Control Program.**

See ITS 3.4.13

* Not required to be performed until 12 hours after establishment of steady state operation. Not applicable to primary-to-secondary leakage.

** Not required to be performed until 12 hours after establishment of steady state operation.

A01

~~TABLE 3.4-6-1~~

~~PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES~~

~~Check Valve No.~~
~~V3227~~
~~V3123~~
~~V3217~~
~~V3113~~
~~V3237~~
~~V3133~~
~~V3247~~
~~V3143~~
~~V3124~~
~~V3114~~
~~V3134~~
~~V3144~~

LA01

Add following valves to list:
V3480, 3481, 3652, 3651 SDC Return Isolation Valves
V3215, 3225, 3235, 3245 SIT Discharge Check Valves

M03

NOTES

SR 3.4.14.1 ~~(a)~~ Maximum Allowable Leakage (each valve):

1. Leakage rates less than or equal to 1.0 gpm are acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount the reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are unacceptable.

(b) ~~To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.~~

LA02

~~(c) Minimum test differential pressure shall not be less than 150 psid.~~

LA03

A01

~~DELETED~~

A01

~~DELETED~~

LCO 3.4.14 2nd Part Applicability

~~EMERGENCY CORE COOLING SYSTEMS~~

The Shutdown Cooling (SDC) System interlock function shall be OPERABLE in MODES 1, 2, 3, and MODE 4 except valves in the SDC flow path when in, or during the transition to or from, the SDC mode of operation.

A03

M04

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
1. V-3659	1. Mini-flow isolation	1. Open
2. V-3660	2. Mini-flow isolation	2. Open

See ITS 3.5.2

b. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.*

2. Verifying ECCS train locations susceptible to gas accumulation are sufficiently filled with water.

. c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. At least once daily of the areas affected within containment by the containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ 267 psia

A03

L05

Add proposed ACTION C

d. In accordance with the Surveillance Frequency Control Program by:

SR 3.4.14.2

1. Verify~~ing proper operation of~~ the open permissive interlock ~~(OPI)~~ and the ~~valve open/~~high SDCS pressure alarm~~s for isolation valves V3651, V3652, V3480, V3481~~.

SR 3.4.14.3

is OPERABLE

LA04

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

See ITS 3.5.2

* Not required to be met for system vent flow paths opened under administrative control.

REACTOR COOLANT SYSTEM

RCS PIV

~~OPERATIONAL~~ LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator (SG),

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

See ITS 3.4.13

LCO 3.4.14
SR 3.4.14.1

e. 1 gpm leakage ~~(except as noted in Table 3.4-1)~~ at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve ~~specified in Table 3.4-1~~.

LA01

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4.

Add proposed Applicability – MODE 4 exception

L02

ACTION:

See ITS 3.4.13

a. With any PRESSURE BOUNDARY LEAKAGE or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System operational leakage greater than any one of the limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE

Add proposed LCO 3.4.14 ACTIONS Note 1

A02

LCO 3.4.14
ACTIONS Note 2

Enter applicable ~~ACTIONS~~ for systems made inoperable by an inoperable pressure isolation valve.

Conditions and Required Actions

Add proposed LCO 3.4.14 Required Action A.1 and A.2 Note

M01

ACTION A

~~c.~~ With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within ~~4~~ hours by use of at least two closed manual or deactivated automatic valves, or be in at least ~~HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

72

L01

ACTION B

MODE 3

36

MODE 5

d. With RCS leakage alarmed and confirmed in a flow path with no flow indication, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:

See ITS 3.4.13

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor in accordance with the Surveillance Frequency Control Program.

b. Monitoring the containment sump inventory and discharge in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

c. *Performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program.

See ITS 3.4.13

d. Monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program.

e. Verifying primary-to-secondary leakage is ≤ 150 gallons per day through any one steam generator in accordance with the Surveillance Frequency Control Program.**

SR 3.4.14.1

~~4.4.6.2.2~~ Each Reactor Coolant System Pressure Isolation Valve check valve ~~specified in Table 3.4-1~~ shall be demonstrated OPERABLE by verifying leakage to be within its limit:

LA01

a. In accordance with the Surveillance Frequency Control Program,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

~~c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,~~

L03

d. Following valve actuation due to automatic or manual action or flow through the valve:

1. Within 24 hours by verifying valve closure, and

2. Within 31 days by verifying leakage rate.

SR 3.4.14.1

~~4.4.6.2.3~~ Each Reactor Coolant System Pressure Isolation Valve motor-operated valve ~~specified in Table 3.4-1~~ shall be demonstrated OPERABLE by verifying leakage to be within its limit;

LA01

a. In accordance with the Surveillance Frequency Control Program, and

~~b. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve.~~

L03

SR 3.4.14.1 Note 1

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

* Not required to be performed until 12 hours after establishment of steady state operation. Not applicable to primary-to-secondary leakage.

See ITS 3.4.13

** Not required to be performed until 12 hours after establishment of steady state operation.

A01

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

Check Valve No.		Motor Operated Valve No.
V3217	V3525	V3480
V3227	V3524	V3481
V3237	V3527	V3652
V3247	V3526	V3651
V3259		
V3258		
V3260		
V3261		
V3215		
V3225		
V3235		
V3245		

LA02

NOTES

SR 3.4.14.1 (a) Maximum Allowable Leakage (each valve):

1. Except as noted below leakage rates greater than 1.0 gpm are unacceptable.

2. For motor-operated valves (MOVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount the reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. For motor-operated valves (MOVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are unacceptable.

(b) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

LA02

(c) Minimum test differential pressure shall not be less than 200 psid.

LA03

A01

~~DELETED~~

A01

~~DELETED~~

A01

~~DELETED~~

LCO 3.4.14 2nd Part Applicability

~~EMERGENCY CORE COOLING SYSTEMS~~

The Shutdown Cooling (SDC) System interlock function shall be OPERABLE in MODES 1, 2, 3, and MODE 4 except valves in the SDC flow path when in, or during the transition to or from, the SDC mode of operation.

A03

M04

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. V3733 V3734	a. SIT Vent Valves	a. Locked Closed
b. V3735 V3736	b. SIT Vent Valves	b. Locked Closed
c. V3737 V3738 V3739 V3740	c. SIT Vent Valves	c. Locked Closed

See ITS 3.5.2

b. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.*

c. In accordance with the Surveillance Frequency Control Program by verifying ECCS locations susceptible to gas accumulation are sufficiently filled with water.

. d. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. At least once daily of the areas affected within containment by the containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

L05

Add proposed ACTION C

e. In accordance with the Surveillance Frequency Control Program by:

SR 3.4.14.3

1. Verifying automatic isolation and interlock action of the shutdown cooling system from Reactor Coolant System when RCS pressure (actual or simulated) is greater than or equal to 515 psia, and that the interlocks prevent opening the shutdown

SR 3.4.14.2

cooling system isolation valves when RCS pressure (actual or simulated) is greater than or equal to 276 psia.

* Not required to be met for system vent flow paths opened under administrative control.

See ITS 3.5.2

DISCUSSION OF CHANGES
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.6.2 Action c specifies the compensatory actions to take when the leakage by any RCS PIV(s) is greater than the specified limit. ITS 3.4.14 ACTIONS A and B also state the appropriate compensatory actions under the same condition, however, ITS 3.4.14 ACTIONS Note 1 has been added. ITS 3.4.14 ACTIONS Note 1 allows separate entry condition for each RCS PIV flow path. This changes the CTS by explicitly stating that the Action is to be taken separately for each inoperable RCS PIV flow path.

The purpose of the Note is to provide explicit instructions for proper application of the ACTION for Technical Specification compliance. In conjunction with proposed Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of the existing Action for inoperable PIVs. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 Unit 1 CTS 4.5.2.d.1 requires verification that proper operation of the open permissive interlock (OPI) and the valve open/high Shutdown Cooling (SDC) System pressure alarms for isolation valves V3651, V3652, V3480, V3481. ITS SR 3.4.14.2 and SR 3.4.14.3 provide similar requirements. ITS SR 3.4.12.2 requires verification that the SDC System interlock prevents the valves from being opened with a simulated or actual RCS pressure signal > 267 psia. ITS SR 3.4.12.3 requires verification that the SDC System interlock function high pressure alarm is OPERABLE. This changes the Unit 1 CTS by clarifying how to verify proper operation of the OPI.

Unit 2 CTS 4.5.2.e.1 requires verifying automatic isolation and interlock action of the shutdown cooling system from Reactor Coolant System when RCS pressure (actual or simulated) is greater than or equal to 515 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when RCS pressure (actual or simulated) is greater than or equal to 276 psia. ITS SR 3.4.14.2 and SR 3.4.14.3 provide similar requirements.

In addition, a new LCO has been added which requires the SDC System interlock function to be OPERABLE. This changes the CTS by including the SDC System interlock Surveillance Requirement with the RCS PIV leakage limits and adding a new LCO and ACTION for the interlock function.

The purpose of CTS Surveillance is to ensure the SDC System interlock function is available to prevent overpressurization of the low pressure SDC System

piping. The OPERABILITY of the low pressure safety injection (LPSI)/SDC system is affected by the position of the SDC return isolation valves, not when the interlock is inoperable. Therefore, the transfer of this requirement to the RCS PIV Specification is appropriate. A discussion of the change to the Applicability of the SDC System interlock (including high pressure alarm for Unit 1) is discussed in DOC M04. A discussion of a change to the Required Actions when the SDC System interlock (including high pressure alarm for Unit 1) is inoperable is discussed in DOC L05. This change is acceptable since the SDC System interlock function is retained in the Technical Specifications. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.4.6.2 Action c. specify the compensatory actions to take when the leakage through any RCS PIV(s) is greater than the specified limit. The compensatory action is to isolate the high pressure portion of the affected system from the low pressure portion of the affected system by use of a combination of at least two closed valves. The CTS does not include any leakage restrictions concerning the valves that may be used to satisfy the isolation requirement of this action. ITS 3.4.14 ACTION A is consistent with the requirement in CTS 3.4.6.2 Action c., however, a Note has been added to the Required Actions (ITS 3.4.14 Required Actions A.1 and A.2 Note) which specifies that each valve used to satisfy ITS 3.4.14 Required Actions A.1 and A.2 must have been verified to meet SR 3.4.14.1, the RCS PIV maximum leakage limit Surveillance Requirement, and either be in the RCS pressure boundary or the high pressure portion of the system. This changes the CTS by providing a Note which explicitly states that the valves used to satisfy Required Action must satisfy the same leakage requirements of the RCS PIVs and provides an option for them to be in the RCS pressure boundary.

The purpose of CTS 3.4.6.2 Action c. is to isolate the flow path in order to minimize the leakage from the high pressure portion of the RCS to the low pressure piping. The ITS 3.4.14 Required Actions A.1 and A.2 Note requires the valves used to provide isolation between the high pressure and low pressure portions of the affected system to have been verified to meet the RCS PIV maximum leakage limits within the required Surveillance Frequency. The addition of the Note represents an additional restriction on unit operation necessary to help ensure the valves used to isolate the high pressure portion from the low pressure portion of the affected system are capable of preventing the overpressurization of the low pressure portion of the system. The ITS 3.4.14 Required Actions A.1 and A.2 Note also provides the option for the valves to be in the RCS pressure boundary, which is considered an acceptable alternative to the high pressure portion of the system. This change is designated as more restrictive because it adds a new requirement to the CTS.

M02 **Unit 1 only:** CTS 4.4.6.2.e provides a verification that the RCS PIV leakage is within limits but provides no Frequency other than “prior to entering MODE 2 after refueling” and “prior to entering MODE 2, whenever the plant has been in COLD SHUTDOWN for 7 days or more if leakage testing has not been

performed in the previous 9 months." ITS SR 3.4.14.1 requires performance on a periodic Frequency or in accordance with the Surveillance Frequency Control Program and within 24 hours following valve actuation due to automatic or manual action or flow through the valve. This changes the CTS by adding a periodic Frequency in accordance with the Surveillance Frequency Control Program and adding the Frequency of within 24 hours following valve actuation due to automatic or manual action or flow through the valve. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per CTS 6.8.4.o. Therefore, SR 3.4.14.1 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 18 months consistent with the ISTS SR 3.4.14.1.

The purpose of CTS 4.4.6.2.e is to perform leakage testing on each RCS PIV to verify that leakage is below the specified limit and to identify each leaking valve. The Frequency in accordance with the Surveillance Frequency Control Program is consistent with 10 CFR 50.55a(g) and the INSERVICE TESTING PROGRAM and is based on the need to perform the Surveillance under conditions that apply during a plant outage. In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. Testing must be performed within 24 hours after the valve has been reseated. Twenty-four hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

This change is designated as more restrictive because it adds new requirements to the CTS.

M03 **Unit 1 only:** CTS Table 3.4.6-1 provides a list of RCS PIVs. ITS 3.4.14 does not contain a list of the RCS PIVs or their associated valve numbers. The list of RCS PIVs in CTS Table 3.4.6-1 are proposed to be relocated to the ITS Bases (See Discussion of Change LA01). The proposed list of RCS PIVs relocated to the ITS Bases includes the shutdown cooling (SDC) return isolation valves and safety injection tank (SIT) discharge check valves consistent with the current licensing basis. In response to NRC Generic Letter 87-06, "Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves," FPL provides testing exception for these additional PIVs (NRC ADAMS Accession No. ML20214W254). The SDC return isolation valves are tested to verify ≤ 1 gpm using indication of differential pressure in isolated downstream piping not to exceed 200 psid. If a leakage rate of > 1 gpm is indicated by > 200 psid, a volumetric leakage test is conducted and verified to be ≤ 50% of the margin between the rate determined by the previous measured leakage rate and 5 gpm. The SIT discharge check valves isolate the SIT from the safety injection header when safety injection header pressure is greater than SIT pressure. SIT pressures and levels are monitored and alarmed. This instrumentation and alarm scheme may indicate possible check valve leakage if it were to occur. As a result, a Note is included in ITS SR 3.4.14.1 (proposed Note 2) to waive performance of the PIV leakage test for the SIT discharge check valves. This change aligns the CTS list of RCS PIVs with the current licensing basis and is designated as more restrictive because it adds additional components to the CTS RCS PIV list.

M04 Unit 1 CTS 4.5.2.d.1 requires, in MODES 1, 2, and 3 with pressurizer pressure ≥ 1750 psia, verification of the proper operation of the open permissive interlock (OPI) and the valve open/high Shutdown Cooling (SDC) System pressure alarms for isolation valves V3651, V3652, V3480, V3481. Unit 2 CTS 4.5.2.e.1 requires, in MODES 1, 2, and 3 with pressurizer pressure ≥ 1750 psia, verifying automatic isolation and interlock action of the shutdown cooling system from Reactor Coolant System when RCS pressure (actual or simulated) is greater than or equal to 515 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when RCS pressure (actual or simulated) is greater than or equal to 276 psia. ITS SR 3.4.14.2 and SR 3.4.14.3 provide similar requirements. However, ITS requires the SDC System interlock function and associated Surveillances to be applicable in MODES 1, 2, and 3, and MODE 4 except valves in the SDC flow path when in, or during the transition to or from, the SDC mode of operation. This changes the CTS by expanding the applicability of the SDC System interlock function to include conditions with pressurizer pressure < 1750 psia.

The purpose of the SDC System interlock function and associated Surveillances is to preclude the SDC return isolation valves from being opened when RCS pressure is above the SDC System design pressure of 350 psig. The SDC System interlock function, along with the PIV leakage requirements support the dominant accident sequence in the intersystem LOCA category of the failure of the low pressure portion of the SDC System outside of containment. Therefore, this change expands the applicability of the interlock function to include MODES where RCS pressure could be greater than SDC System design pressure. This change is consistent with the ISTS Applicability for RCS PIVs and is designated as more restrictive because it expands the CTS applicability for this interlock function.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.4.6.2.e, 4.4.6.2.e require each RCS PIV leakage be within the limits for the RCS PIVs listed in Table 3.4.6-1. Unit 2 CTS 3.4.6.2.e, 4.4.6.2.2 and 4.6.2.2.3 require each RCS PIV leakage to be within the limits for the RCS PIVs listed in Table 3.4-1. ITS 3.4.14 does not contain a list of the RCS PIVs or their associated valve numbers. This changes the CTS by relocating the list of RCS PIVs and their associated valve numbers to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.4.14 requires the RCS PIVs to be OPERABLE, and ITS SR 3.4.14.1 requires periodic Surveillances to determine RCS PIV leakage. It is not necessary for the list of RCS PIVs to be in the Technical Specifications in order to ensure that the RCS PIVs are OPERABLE.

Other lists of components, such as containment isolation valves and equipment response time, have been relocated from the Technical Specification to licensee-controlled documents while retaining the requirements on these components in Technical Specifications. Also, this change is acceptable because these types of description details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system description is being removed from the Technical Specifications.

LA02 *(Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS Table 3.4.6-1 and Unit 2 CTS Table 3.4-1 are modified by Note (b). Note (b) explains an alternative method of testing the PIVs to satisfy the ALARA requirements. ITS 3.4.14 does not retain this Note. This changes the CTS by relocating the information in the Note to the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specification is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.4.14 still retains the requirements that RCS PIV leakage must be within limit and provides the appropriate Surveillance that includes the leakage limit. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS Table 3.4.6-1 and Unit 2 CTS Table 3.4-1 are modified by Note (c). Note (c) specifies the minimum test differential pressure for the RCS PIVs to not be below (150 psid on Unit 1, 200 psid on Unit 2). ITS 3.4.14 does not specify this limit. This changes the CTS by relocating the RCS PIV minimum test differential pressure to the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.4.14 retains the requirement that the RCS PIV leakage must be within limit and provides the appropriate Surveillance that includes the leakage limit. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA04 **Unit 1 only:** *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 4.5.2.d.1 requires, in part, verification that the valve open/high Shutdown Cooling (SDC) System pressure alarms for isolation valves V3651, V3652, V3480, V3481. ITS SR 3.4.12.3 requires verification that the SDC System interlock function high pressure alarm is OPERABLE. This changes the CTS by relocating detail related to the specific valve numbers associated with the RCS high pressure alarm that is actuated when the SDC return isolation valves are open.

The removal of these details, which are related to detail system design and description, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS LCO 3.4.14 requires the SDC System interlock function of be OPERABLE and ITS SR 3.4.14.3 requires periodic verification that the SDC System interlock function high pressure alarm is OPERABLE. It is not necessary to state the valve numbers associated with the alarm function in the Technical Specifications in order to ensure that the SDC System interlock function high pressure alarm is OPERABLE. This change is acceptable because these type of description details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system description is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 - Relaxation of Completion Time)* CTS 3.4.6.2 Action c. requires, in part, that if the RCS PIV leakage is not within limits, the high pressure to low pressure boundary must be isolated by at least two valves as specified in CTS 3.4.6.2 Action c. Specifically, Unit 1 CTS 3.4.6.2 Action c. states, in part, that with the integrity of any pressure isolation valve specified in Table 3.4.6-1 not demonstrated, power operation may continue provided at least two valves in each high pressure line that has a non-functional valve are in and remain in, the mode corresponding to the isolated condition. Unit 2 CTS 3.4.6.2 Action c. states, in part, that with the integrity of any pressure isolation valve specified in Table 3.4-1 not demonstrated, power operation may continue provided that the high pressure portion of the affected system is isolated from the low pressure portion by use of at least two closed manual or deactivated automatic valves. The two CTS Actions ultimately result in requiring the two valves to be in the isolated condition within 4 hours. ITS 3.4.14 ACTION A contains similar requirements but allows 4 hours to isolate the first valve (Required Action A.1) and 72 hours to isolate the second valve (Required Action A.2). This changes the CTS by extending the time requirement to close the second valve from 4 hours to 72 hours.

The purpose of CTS 3.4.6.2 Action c is to allow time to reduce leakage before isolating the pathway. This change is acceptable because the Completion Time

is consistent with safe operation under the specified Condition, considering the low probability of a DBA occurring during the allowed Completion Time. This includes the fact that the high pressure to low pressure boundary is isolated by at least one isolation device. The time to close the first valve remains the same and the time to close the second valve has been changed from 4 hours to 72 hours. The 4 hour Completion Time to close the first valve ensures leakage in excess of the allowable limit is reduced. The 4 hour time allows time for these actions and restricts the time of operation with leaking valves. The 72 hours Completion Time to close the second valve considers the time required to complete the Required Action and the low probability of the first valve failing during this period. This change is designated as less restrictive because additional time is allowed to perform a required action than was allowed in the CTS.

L02 *(Category 2 - Relaxation of Applicability)* CTS 3.4.6.2 is applicable in MODES 1, 2, 3, and 4. ITS 3.4.14 is applicable in MODES 1, 2, 3, and MODE 4, except valves in the shutdown cooling (SDC) flow path when in, or during the transition to or from, the SD mode of operation. This changes the CTS by exempting the SDC flow path PIVs from the leakage requirements when in or during the transition to or from the SDC mode of operation. | R1

The purpose of CTS 3.4.6.2.e is to ensure the RCS PIVs are within leakage limits. This change is acceptable because the LCO requirements continue to ensure that the components are maintained consistent with the safety analyses and licensing basis. It is not necessary for the SDC PIVs to meet the leakage limits when in or during transition to or from the SDC mode of operation. These check valves cannot open until the SDC system is placed in service, which is not until RCS pressure is less than the SDC permissive. Thus, overpressurization of the SDC piping is not a concern. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than are being applied in the CTS.

L03 *(Category 5 - Deletion of Surveillance Requirement)* Unit 1 CTS 4.4.6.2.e.3 and Unit 2 CTS 4.4.6.2.2.c and 4.4.6.2.3.b require testing of RCS PIVs following maintenance, repair, or replacement work on the valve. ITS 3.4.14 does not include this requirement. This changes the CTS by eliminating a post-maintenance Surveillance Requirement. | R1

The purpose of the Surveillance requirements is to ensure the RCS PIV leakage is within limits prior to returning the valve to service. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Whenever, the OPERABILITY of a system or component has been affected by repair, maintenance, modification, or replacement of a component, post maintenance testing is required to demonstrate the OPERABILITY of a system or component. This is described in the Bases for ITS SR 3.0.1 and required under SR 3.0.1. In addition, the requirements of 10 CFR 50, Appendix B, Section XI (Test Control), provide adequate controls for test programs to ensure that testing incorporates applicable acceptance criteria. Compliance with 10 CFR 50, Appendix B is required under the unit operating license. As a result,

post-maintenance testing will continue to be performed and an explicit requirement in the Technical Specifications is not necessary. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L04 **Unit 1 only:** *(Category 5 - Deletion of Surveillance Requirement)* CTS 4.4.6.2.f provides additional compensatory measures to take, above those required by CTS 3.6.4.2 Action c., when leakage through an RCS PIV is not within limit. The CTS requires a daily leakage test of the remaining OPERABLE RCS PIV in the flow path or a combined leakage test of the two valves used to comply with CTS 3.6.4.2 Action c. In addition, the position of the second, non-RCS PIV valve is required to be recorded on a daily basis. ITS 3.4.14 does not include these additional compensatory measures. This changes the CTS by deleting the additional Surveillance compensatory measures taken when leakage through an RCS PIV is not within limit.

The purpose of CTS 4.4.6.2.f is to help ensure that the leakage through the valves used to isolate the penetration with an inoperable RCS PIV is minimized so that an overpressurization event of the downstream piping cannot occur. The change is acceptable since the requirements to ensure the leakage through the two closed valves is within the RCS PIV leakage limit and to ensure closure of the valves are maintained in the ITS. The RCS PIV leakage is ensured prior to using each of the valves as an isolation boundary, as required by the ITS 3.4.14 Required Action A.1 and A.2 Note. Once leakage is checked, it is not expected to change since the valve cannot be manipulated (ITS 3.4.14 ACTION A requires the valves to be isolated, thus they must remain isolated to comply with the ACTION). Manipulation of manual valves that have been closed and automatic valves that have been deactivated to comply with Technical Specification Actions is a controlled evolution and the valves are not expected to be inadvertently moved from the isolated condition. Furthermore, these valves will be verified to be in the correct position when first isolated to comply with ITS 3.4.14 ACTIONS A and B. This change is designated as less restrictive because a Surveillance compensatory measure required by the CTS will not be required in the ITS.

L05 (*Category 4 – Relaxation of Required Action*) Unit 1 CTS 4.5.2.d.1 requires verification that proper operation of the open permissive interlock (OPI) and the valve open/high SDC System pressure alarms for isolation valves V3651, V3652, V3480, V3481. Unit 2 CTS 4.5.2.e.1 requires verifying automatic isolation and interlock action of the shutdown cooling system from Reactor Coolant System when RCS pressure (actual or simulated) is greater than or equal to 515 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when RCS pressure (actual or simulated) is greater than or equal to 276 psia. When the interlock is inoperable (i.e., the Surveillance not met), LCO 3.0.3 entry is required since this inoperability affects both LPSI trains. ITS 3.4.14 ACTION C requires the isolation of the penetration by use of one closed manual or deactivated automatic valve within 4 hours. This changes the CTS by allowing the penetration to be isolated and to continue operation of the unit for an unlimited amount of time without entry into LCO 3.0.3.

At PSL, the LPSI and SDC systems share the same components. The purpose of ITS 3.4.14 ACTION C is to isolate the penetration to ensure the LPSI/SDC

System is not overpressurized by the RCS. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded condition in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the action of isolating the penetration by use of at least one closed manual or deactivated automatic valve accomplishes the function of ensuring the valves are not inadvertently opened when RCS pressure is above the SDC System design pressure. The 4-hour Completion Time is reasonable based on the consideration that the associated valves are closed when the SDC System is not aligned for the SDC function and the low probability of a failure that could inadvertently open the isolation valves. This change allows the unit to continue to operate and avoids an unnecessary plant transient as a result of entry into LCO 3.0.3. Closing and deactivating an automatic valve or closing a manual valve will ensure the function of the interlock is met. Therefore, the actions provide sufficient remedial measures to preclude overpressurization of the LPSI/SDC System by the RCS thereby allowing continued safe operation pursuant the requirements of 10 CFR 50.36(c)(2). This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

3.4.6.2.e

LCO 3.4.14 Leakage from each RCS PIV shall be within limits.

DOC A03

AND

The Shutdown Cooling (SDC) System interlock function shall be OPERABLE. (1)

3.4.6.2 Applicability, DOC M04 DOC L02

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the ~~shutdown cooling (~~SDC~~)~~ flow path when in, or during the transition to or from, the SDC mode of operation. (1)

ACTIONS

----------NOTES----------

DOC A02

1. Separate Condition entry is allowed for each flow path.

3.4.6.2 Action c. Note

2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.6.2 Action c., DOC M01 A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	----------NOTE---------- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.1 and be on the RCS pressure boundary ~~[~~or the high pressure portion of the system~~]~~. (2) --- A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve. AND	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.6.2 Action c.	A.2 ~~[~~Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve. ~~[or]~~ ~~Restore RCS PIV to within limits.~~	72 hours ~~]~~ (2) (2)
3.4.6.2 Action c. B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
DOC L05 C. ~~[~~Shutdown Cooling (SDC) System ~~autoclosure~~ interlock function inoperable.	C.1 Isolate the affected penetration by use of one closed manual or deactivated automatic valve.	4 hours ~~]~~ (2)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.14.1 ----------NOTES---------- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on ~~the RCS PIVs located in the SDC flow path when in the shutdown cooling mode of operation.~~ (safety injection tank discharge check valves.) ~~3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided~~. ---------- Verify leakage from each RCS PIV is equivalent to ≤ ~~0.5~~ 5.0 gpm ~~per nominal inch of valve size up to a maximum of 5 gpm~~ at an RCS pressure ≥ ~~[2215]~~ 2225 psia and ≤ ~~[2255]~~ 2275 psia; and when current measured rate is > 1 gpm, the current measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and 5.0 gpm by 50%	~~In accordance with the INSERVICE TESTING PROGRAM, and [[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ AND Prior to entering MODE 2 determine the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months AND

4.4.6.2.e
4.4.6.2.e.4
4.4.6.2.e
Table 3.4.6-1 Note (a)
DOC M02
4.4.6.2.e.2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
DOC M02		Within 24 hours following valve actuation due to automatic or manual action or flow through the valve	
4.5.2.d.1	SR 3.4.14.2	~~NOTE~~ ~~[Not required to be met when the SDC System autoclosure interlock is disabled in accordance with SR 3.4.12.7.~~ Verify SDC System ~~autoclosure~~ interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ~~≥ [425] psig~~. (> 267 psia)	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~	8 3 2 3 2
4.5.2.d.1	SR 3.4.14.3	~~NOTE~~ ~~[Not required to be met when the SDC System autoclosure interlock is disabled in accordance with SR 3.4.12.7.~~ Verify SDC System ~~autoclosure~~ interlock ~~causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ [600] psig~~. (function high pressure alarm is OPERABLE.)	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~	8 3 2 3 2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

3.4.6.2.e

LCO 3.4.14 Leakage from each RCS PIV shall be within limits.

DOC A03

AND

The Shutdown Cooling (SDC) System interlock function shall be OPERABLE. 1

3.4.6.2 Applicability, DOC M04 DOC L02

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the ~~shutdown cooling (~~SDC~~)~~ flow path when in, or during the transition to or from, the SDC mode of operation. 1

ACTIONS

----------NOTES----------

DOC A02

1. Separate Condition entry is allowed for each flow path.

3.4.6.2 Action c. Note

2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.6.2 Action c., DOC M01 A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	----------NOTE---------- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.1 and be on the RCS pressure boundary ~~[~~or the high pressure portion of the system~~]~~. ---------- A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve. AND	4 hours

2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2 [Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve. [or] Restore RCS PIV to within limits.	72 hours]
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. [Shutdown Cooling (SDC) System autoclosure interlock function inoperable.	C.1 Isolate the affected penetration by use of one closed manual or deactivated automatic valve.	4 hours]

3.4.6.2 Action c.

3.4.6.2 Action c.

DOC L05

2

2

2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	----------NOTES----------	
	1. Not required to be performed in MODES 3 and 4.	
	2. Not required to be performed on the RCS PIVs located in the SDC flow path when in the shutdown cooling mode of operation.	
	3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided.	

	Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ [2215] psia and ≤ [2255] psia.	In accordance with the INSERVICE TESTING PROGRAM, and [[18] months OR In accordance with the Surveillance Frequency Control Program] AND Prior to entering MODE 2 determine the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months AND

4.4.6.2.2
4.4.6.2.3 Note

4.4.6.2.2
Table 3.4-1, Note (a)

4.4.6.2.2.a

4.4.6.2.2.b

5.0

2225

2275

; and when current measured rate is > 1 gpm, the current measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and 5.0 gpm by 50%

4 5 4 5 6 3 2 3 2 3

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.6.2.2.d			Within 24 hours following valve actuation due to automatic or manual action or flow through the valve	
4.5.2.e.1	SR 3.4.14.2	~~NOTE~~ ~~[Not required to be met when the SDC System autoclosure interlock is disabled in accordance with SR 3.4.12.7.~~		8
		Verify SDC System ~~autoclosure~~ interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ ~~[425] psig~~. (276 psia)	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~	3 2 3 2
4.5.2.e.1	SR 3.4.14.3	~~NOTE~~ ~~[Not required to be met when the SDC System autoclosure interlock is disabled in accordance with SR 3.4.12.7.~~		8
		Verify SDC System ~~autoclosure~~ interlock causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ ~~[600] psig~~. (515 psia)	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~	3 2 3 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

1. The second part of the LCO has been added to ensure consistency between the LCO, ACTIONS, and Surveillance Requirements. The ISTS LCO, Actions, and Surveillances do not match up since there is no explicit statement in the LCO requiring the Shutdown Cooling (SDC) System interlock function to be OPERABLE. In the ISTS, if the SDC System interlock function is inoperable, the LCO is not affected since RCS PIV leakage is not directly affected by an inoperable interlock function. Therefore, the inclusion of the second portion of the LCO ensures consistency between the LCO, ACTIONS, and Surveillance Requirements. In addition, due to the addition of the term "SDC" into the LCO statement, the use of the term " Shutdown Cooling (SDC)" in the Applicability has been changed to "SDC."

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

4. Note 2 to ISTS SR 3.4.14.1 has been deleted since it is not necessary. The ISTS 3.4.14 Applicability does not require leakage to be met for SDC valves in the flow path when in MODE 4 and when in, or during the transition to or from, the SDC Mode of operation.

5. Note 3 to ISTS SR 3.4.14.1 has been deleted since it is not required by the current licensing basis.

6. Verification of RCS PIV leakage is required by the INSERVICE TESTING PROGRAM, which fulfills the requirements of 10 CFR 50.55a(g). Therefore, it is duplicative to require ISTS SR 3.4.14.1 to be performed in accordance with the INSERVICE TESTING PROGRAM. Since compliance with 10 CFR 50.55a is required by the operating licenses of PSL Unit 1 and Unit 2, this Frequency is not included in ITS SR 3.4.14.1.

7. **Unit 1 only:** A Note is added to ISTS 3.4.14.1 (Note 2 to ITS SR 3.4.14.1). This Note waives performance of the PIV leakage test for the safety injection tank (SIT) discharge check valves. The SIT discharge check valves are added to the list of RCS PIVs consistent with the current licensing basis. In response to NRC Generic Letter 87-06, “Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves,” an exception to PIV testing was provided for the SIT discharge check valves (NRC ADAMS Accession No. ML20214W254). The SIT discharge check valves isolate the SIT from the safety injection header when safety injection header pressure is greater than SIT pressure. SIT pressures and levels are monitored and alarmed. This instrumentation and alarm scheme may indicate possible check valve leakage if it were to occur. As a result, periodic performance of the leakage testing is not required for the SIT discharge check valves consistent with current licensing basis.

8. The Note to ISTS SRs 3.4.14.2 and 3.4.14.3 is deleted and not included in the ITS because ISTS SR 3.4.12.7 is not related to SRs 3.4.14.2 and 3.4.14.3. The Note to

the SRs is also included in NUREG-1431, “Standard Technical Specifications, Westinghouse Plants,” and refers to an SR that requires the associated residual heat removal (RHR) suction isolation valve to be locked open with operator power removed for each required RHR suction relief valve. There is no equivalent SR in NUREG-1432 or the PSL Unit 1 and Unit 2 current Technical Specifications. Therefore, the Note is not included in the PSL Unit 1 and Unit 2 ITS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the RCS pressure boundary that separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both PIVs in series in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two valves in series is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed condition that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

BASES

BACKGROUND (continued)

a. Shutdown Cooling (SDC) System,

b. Safety Injection System, and

c. Chemical and Volume Control System.

The PIVs are listed in Table B 3.4.14-1 ~~FSAR section (Ref. 6)~~. 1

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier. Insert 1 1

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the SDC System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the reactor coolant pressure boundary (RCPB), and the subsequent pressurization of the SDC System downstream of the PIVs from the RCS. Because the low pressure portion of the SDC System is ~~typically~~ designed for ~~[600]~~ psig, overpressurization failure of the SDC low pressure line would result in a LOCA outside containment and subsequent risk of core melt. 350 2

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is ~~0.5 gpm per nominal inch of valve size, with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.~~ 1

< 5.0 gpm. However, when the current measured rate is > 1.0 gpm, the current measured rate shall not exceed the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate (5.0 gpm) by 50%. 1

1

1

INSERT 1

Two motor operated valves are included in series in the suction piping of the two SDC trains to isolate the high pressure RCS from the low pressure piping of the SDC System when the RCS pressure is above the design pressure of the SDC System piping and components. Ensuring the SDC interlock that prevents the valves from being opened is OPERABLE ensures that RCS pressure will not pressurize the SDC System beyond its design pressure of 350 psig.

BASES

LCO (continued)

Ensuring the SDC System interlock function is OPERABLE, including the open permissive interlock that prevents the valves from being opened and the high pressure alarm on RCS high pressure when the valves are open, ensures that RCS pressure will not pressurize the SDC System beyond its design pressure.

6

Reference ~~7~~ permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

1

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the SDC flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the SDC mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based on the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note stating that the valves used for isolation must meet the same leakage requirements as the PIVs and must be in the RCPB ~~[~~or the high pressure portion of the system~~]~~.

2

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate if leakage cannot be reduced. The 4 hours allows the actions and restricts the operation with leaking isolation valves.

BASES

ACTIONS (continued)

[Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the action and the low probability of a second valve failing during this time period.

~~or~~

~~The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period.]~~

~~REVIEWER'S NOTE~~
~~Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.~~

B.1 and B.2

any Required Action of Condition A cannot be met within the required Completion Time

If ~~leakage cannot be reduced [,the system isolated,] or other Required Actions accomplished~~, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This Action reduces the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

or incapable of alerting the operator of increasing RCS pressure with the SDC return isolation valves open

The inoperability of the SDC ~~autoclosure~~ interlock renders the SDC suction isolation valves incapable of~~: isolating in response to a high pressure condition and~~ preventing inadvertent opening of the valves at RCS pressures in excess of the SDC systems design pressure. If the SDC ~~autoclosure~~ interlock is inoperable, operation may continue as long as the affected SDC suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the ~~autoclosure~~ function.

The RCS PIV leakage limit is < 5.0 gpm. However, RCS PIV leakage is also limited when the current measured rate is > 1.0 gpm, such that the current measured rate shall not exceed the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and 5.0 gpm by 50%. The minimum differential test pressure across each valve shall be ≥ 150 psid.

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 or A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. ~~The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve.~~ Leakage testing requires a stable pressure condition.

1

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 9 months, but may be extended if the plant does not go into MODE 5 for at least 7 days. ~~[The [18] month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) and the INSERVICE TESTING PROGRAM and is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref. 7), and is based on the need to perform the Surveillance under conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~----------]~~

2

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. ~~PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided.~~ Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

1

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by two Notes. Note 1 states that the Surveillance is not required to be performed in MODES 3 and 4.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

3

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. ~~The Note that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the SDC System when the SDC System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the SDC shutdown cooling flow path must be leakage rate tested after SDC is secured and stable unit conditions and the necessary differential pressures are established.~~

3

Insert 2

1

SR 3.4.14.2 and SR 3.4.14.3

function, which includes the open permissive interlock and the high pressure alarm, is

Verifying that the SDC ~~autoclosure~~ interlock~~s~~ are OPERABLE ensures that RCS pressure will not pressurize the SDC system beyond ~~125% of~~ its design pressure of ~~[600]~~ psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be ~~< [425] psig~~ to open the valves. This setpoint ensures the SDC design pressure will not be exceeded and the SDC relief valves will not lift. ~~[The 18 month Frequency is based on the need to perform these Surveillances under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.~~

350

≤ 267 psia

Verifying the SDC interlock function high pressure alarm is OPERABLE ensures that when RCS pressure exceeds the high pressure setpoint and one or more SDC return isolation valves (i.e., V3651, V3652, V3480, V3481) are open, an alarm will annunciate in the control room to alert the operator.

1

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

2

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

1

1

INSERT 2

Note 2 modifies this SR to waive performance of the PIV leakage test for the safety injection tank (SIT) discharge check valves. In response to NRC Generic Letter 87-06, an exception to PIV leakage testing was provided for the SIT discharge check valves (Ref. 7). SIT pressures and levels are monitored and alarmed. The SIT pressure and level instrumentation and alarm scheme may be used to indicate possible check valve leakage if it were to occur. As a result, periodic performance of the leakage testing for the SIT discharge check valves is not required.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The SRs are modified by Notes allowing the SDC autoclosure function to be disabled when using the SDC System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.~~

1

REFERENCES

1. 10 CFR 50.2.

2. 10 CFR 50.55a(c).

3. 10 CFR 50, Appendix A, Section V, GDC 55.

4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.

5. NUREG-0677, May 1980.

~~6. [Document containing list of PIVs.]~~

2

6

~~7~~. ASME Code for Operation and Maintenance of Nuclear Power Plants.

~~8. 10 CFR 50.55a(g).~~

1

7. Letter from C. Woody (FPL) to Document Control Desk (NRC), "Generic Letter 87-06, "Periodic Verification of Leaktight Integrity of Pressure Isolation Valves," dated June 10, 1987 (NRC ADAMS Accession No. ML20214W254).

Insert 3

1

INSERT 3

Table B 3.4.14-1 (Page 1 of 1)
Reactor Coolant System Pressure Isolation Valves

CHECK VALVE NUMBER	MOTOR OPERATED VALVE NUMBER
V3217	V3480
V3227	V3481
V3237	V3652
V3247	V3651
V3113	
V3123	
V3133	
V3143	
V3114	
V3124	
V3134	
V3144	
V3215	
V3225	
V3235	
V3245	

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the RCS pressure boundary that separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both PIVs in series in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two valves in series is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed condition that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

BASES

BACKGROUND (continued)

a. Shutdown Cooling (SDC) System,

b. Safety Injection System, and

c. Chemical and Volume Control System.

The PIVs are listed in ~~FSAR section (Ref. 6)~~ Table B 3.4.14-1. 1

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier. Insert 1 1

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the SDC System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the reactor coolant pressure boundary (RCPB), and the subsequent pressurization of the SDC System downstream of the PIVs from the RCS. Because the low pressure portion of the SDC System is ~~typically~~ designed for ~~[600]~~ psig, overpressurization failure of the SDC low pressure line would result in a LOCA outside containment and subsequent risk of core melt. 350 2

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is ~~0.5 gpm per nominal inch of valve size, with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.~~ 1

< 5.0 gpm. However, when the current measured rate is > 1.0 gpm, the current measured rate shall not exceed the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate (5.0 gpm) by 50%. 1

1

1

INSERT 1

Two motor operated valves are included in series in the suction piping of the two SDC trains to isolate the high pressure RCS from the low pressure piping of the SDC System when the RCS pressure is above the design pressure of the SDC System piping and components. Ensuring the SDC interlock that prevents the valves from being opened is OPERABLE ensures that RCS pressure will not pressurize the SDC System beyond its design pressure of 350 psig.

Ensuring the SDC System interlock function is OPERABLE ensures that RCS pressure will not pressurize the SDC System beyond its design pressure by providing an open permissive interlock. The SDC System interlock function also automatically closes the SDC suction line isolation valves on high pressurizer pressure with a setpoint high enough to preclude premature isolation of the SDC System. 1

BASES

LCO (continued)

6

Reference ~~7~~ permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power. 1

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the SDC flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the SDC mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based on the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note stating that the valves used for isolation must meet the same leakage requirements as the PIVs and must be in the RCPB ~~[~~or the high pressure portion of the system~~]~~. 2

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate if leakage cannot be reduced. The 4 hours allows the actions and restricts the operation with leaking isolation valves.

BASES

ACTIONS (continued)

[Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the action and the low probability of a second valve failing during this time period. 2

~~or~~

~~The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period.]~~ 2

~~REVIEWER'S NOTE~~
~~Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.~~ 1

B.1 and B.2

any Required Action of Condition A cannot be met within the required Completion Time

If ~~leakage cannot be reduced [the system isolated] or other Required Actions accomplished~~, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This Action reduces the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. 3

C.1

The inoperability of the SDC ~~autoclosure~~ interlock renders the SDC suction isolation valves incapable of: isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the SDC systems design pressure. If the SDC ~~autoclosure~~ interlock is inoperable, operation may continue as long as the affected SDC suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the ~~autoclosure~~ function. 1

The RCS PIV leakage limit is < 5.0 gpm. However, RCS PIV leakage is also limited when the current measured rate is > 1.0 gpm, such that the current measured rate shall not exceed the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and 5.0 gpm by 50%. The minimum differential test pressure across each valve shall be ≥ 200 psid.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 or A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. ~~The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve.~~ Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 9 months, but may be extended if the plant does not go into MODE 5 for at least 7 days. ~~[The [18] month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) and the INSERVICE TESTING PROGRAM and is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref. 7), and is based on the need to perform the Surveillance under conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~----------]~~

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. ~~PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided.~~ Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note that states the Surveillance is not required to be performed in MODES 3 and 4.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

3

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. ~~The Note that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the SDC System when the SDC System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the SDC shutdown cooling flow path must be leakage rate tested after SDC is secured and stable unit conditions and the necessary differential pressures are established.~~

3

SR 3.4.14.2 and SR 3.4.14.3

automatic closure of the suction isolation valves

System while precluding premature isolation of the system

276 psia

System design pressure of 350 psig

System suction

Verifying ~~that~~ the SDC ~~autoclosure~~ interlock~~s are~~ ~~OPERABLE~~ ensures that RCS pressure will not pressurize the SDC ~~system beyond 125% of its design pressure of [600] psig~~. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < ~~[425] psig~~ to open the valves. This setpoint ensures the SDC ~~design pressure~~ will not be exceeded and the SDC relief valves will not lift. ~~[The 18 month Frequency is based on the need to perform these Surveillances under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.~~

1

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

2

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~The SRs are modified by Notes allowing the SDC autoclosure function to be disabled when using the SDC System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.~~ 1

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
~~6. [Document containing list of PIVs.]~~ 2
6
~~7~~. ASME Code for Operation and Maintenance of Nuclear Power Plants. 1
~~8. 10 CFR 50.55a(g).~~

Insert 2

1

INSERT 2

Table B 3.4.14-1 (Page 1 of 1)
Reactor Coolant System Pressure Isolation Valves

CHECK VALVE NUMBER	MOTOR OPERATED VALVE NUMBER
V3217	V3480
V3227	V3481
V3237	V3652
V3247	V3651
V3259	
V3258	
V3260	
V3261	
V3215	
V3225	
V3235	
V3245	
V3525	
V3524	
V3527	
V3526	

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.14 BASES, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to be consistent with changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.14, RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 15

3.4.15, RCS Leakage Detection Instrumentation

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

instrumentation shall

LCO 3.4.15 ~~3.4.6.1~~ The following RCS leakage detection ~~systems~~ will be OPERABLE:

One

LCO 3.4.15.a ~~a.~~ ~~The~~ reactor cavity sump inlet flow monitor~~ing system; and~~

LCO 3.4.15.b ~~b.~~ One containment atmosphere radioactivity monitor (gaseous or particulate).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Condition A ~~a.~~ ~~With the~~ reactor cavity sump inlet flow monitor~~ing system~~ inoperable ~~with an operable containment particulate radioactivity monitor~~, perform a ~~RCS water inventory balance~~ at least once per 24* hours and restore the sump inlet flow monitor~~ing system~~ to OPERABLE status within 30 days; ~~otherwise,~~ be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

Required Action A.1 — SR 3.4.13.1 — A02

Required Action A.2

ACTION D — MODE 3 — MODE 5 — 36

AND containment atmosphere particulate radioactivity monitor inoperable

Condition C ~~b.~~ ~~With the~~ reactor cavity sump inlet flow monitor~~ing system~~ inoperable ~~with only the containment gaseous radioactivity monitor operable~~, perform ~~an RCS water inventory balance~~ at least once per 24* hours and analyze grab samples of the containment atmosphere at least once per 12 hours, and either restore the sump inlet flow monitor~~ing system~~ to OPERABLE status within 7 days or restore the containment particulate radioactivity monitor to OPERABLE status within 7 days ~~and enter action a. above with the time in this action applied against the allowed outage time of action a.; otherwise,~~ be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

Required Action A.1 — SR 3.4.13.1 — A02

Required Action C.1

Required Action C.2

Condition C Note — A03

ACTION D — MODE 3 — MODE 5 — 36

Condition B ~~c.~~ With the required radioactivity monitor inoperable, analyze grab samples of the containment atmosphere or perform ~~a RCS water inventory balance~~ at least once per 24* hours, and restore the required radioactivity monitor to OPERABLE status within 30 days; ~~otherwise,~~ be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

Required Action B.1.1 — SR 3.4.13.1 — A02

Required Action B.1.2

Required Action B.2

ACTION D — MODE 3 — MODE 5 — 36

ACTION E ~~d.~~ ~~With~~ all required monitors inoperable, enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

~~4.4.6.1 The RCS leakage detection instruments shall be demonstrated OPERABLE by:~~

SR 3.4.15.1, SR 3.4.15.2 SR 3.4.15.4 ~~a.~~ Perform~~ance of the~~ CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor in accordance with ~~Surveillance Requirement 4.3.3.1~~ the Surveillance Frequency Control Program. — A02

SR 3.4.15.3 ~~b.~~ Perform~~ance of the~~ CHANNEL CALIBRATION of the reactor cavity sump inlet flow monitor~~ing system~~ in accordance with the Surveillance Frequency Control Program.

Required Action A.1 Note, B.1.2 Note, * Not required to be performed until 12 hours after establishment of steady state operation.

~~DELETED~~

~~INSTRUMENTATION~~ ← REACTOR COOLANT SYSTEMS (RCS)

RCS Leakage Detection

3/4.3.3 ~~MONITORING~~ INSTRUMENTATION

A04

~~RADIATION MONITORING~~

LIMITING CONDITION FOR OPERATION

(gaseous or particulate)

One containment

LCO 3.4.15.b 3.3.3.1 ~~The~~ radiation monitor~~ing instrumentation~~ channel~~s~~ ~~shown in Table 3.3-6~~ shall be OPERABLE ~~with their alarm/trip setpoints within the specified limits~~.

Applicability **APPLICABILITY**: ~~As shown in Table 3.3-6.~~ ← MODES 1, 2, 3, and 4

ACTION:

See ITS 3.3.6
ITS 3.3.7
ITS 3.3.8
ITS 3.3.9

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

A04

required

ACTIONS b With one or more ~~radiation monitoring~~ channel~~s~~ inoperable, take the ACTION shown in ~~Table 3.3-6.~~ ← Specification 3.4.15

~~c. The provisions of Specification 3.0.3 are not applicable.~~

A05

SURVEILLANCE REQUIREMENTS

required containment

SR 3.4.15.1
SR 3.4.15.2
SR 3.4.15.4 4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program.

A04

4.3.3.2 In accordance with the Surveillance Frequency Control Program, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

See ITS 3.3.7

~~TABLE 3.3-6~~

~~RADIATION MONITORING INSTRUMENTATION~~

~~INSTRUMENT~~	~~MINIMUM CHANNELS OPERABLE~~	~~APPLICABLE MODES~~	~~ALARM/TRIP SETPOINT~~	~~MEASUREMENT RANGE~~	ACTION	
~~1. AREA MONITORS~~						
a. Fuel Storage Pool Area	1	*	≤ 15 mR/hr	10^{-1} – 10^{4} mR/hr	13	See ITS 3.3.8
b. Containment (CIS)	3	****	≤ 90 mR/hr	1 – 10^{5} mR/hr	16	See ITS 3.3.6
c. Containment Area – Hi Range	1	1, 2, 3, & 4	≤ 10 R/hr	1 – 10^{7} R/hr	15	See ITS 3.3.9
d. Control Room Isolation	1 per intake	ALL MODES	≤ 320 cpm	10 - 10^{7} cpm	17	See ITS 3.3.7
2. PROCESS MONITORS						
LCO 3.4.15 a. Containment						
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	~~Not Applicable~~	~~10 – 10^{6} cpm~~	14	
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	~~Not Applicable~~	~~10 – 10^{6} cpm~~	14	

A06 or

LA01

* With fuel in the storage pool or building. See ITS 3.3.8

**** During movement of recently irradiated fuel assemblies within containment. See ITS 3.3.6

~~TABLE 3.3-6 (Continued)~~

~~TABLE NOTATION~~

ACTION 12 - DELETED

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTIONS

~~ACTION 14~~ - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification ~~3.4.6.1~~. ← 3.4.15

A06

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s),and

2) Prepare and submit a Special Report to the Commission pursuant to Specification6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

See ITS 3.3.9

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

A01

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

instrumentation shall

LCO 3.4.15 ~~3.4.6.1~~ The following RCS leakage detection ~~systems~~ ~~will~~ be OPERABLE:

One

LCO 3.4.15.a ~~a.~~ ~~The~~reactor cavity sump inlet flow monitor~~ing system; and~~

LCO 3.4.15.b ~~b.~~ One containment atmosphere radioactivity monitor (gaseous or particulate).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Condition A ~~a.~~ ~~With the~~ reactor cavity sump inlet flow monitor~~ing system~~ inoperable ~~with an operable containment particulate radioactivity monitor~~, perform a ~~RCS water inventory balance~~ at least once per 24* hours and restore the sump inlet flow monitor~~ing system~~ to OPERABLE status within 30 days; ~~otherwise,~~ be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

SR 3.4.13.1 A02

Required Action A.1

Required Action A.2

ACTION D

MODE 3

36

MODE 5

AND containment atmosphere particulate radioactivity monitor inoperable

Condition C ~~b.~~ ~~With the~~ reactor cavity sump inlet flow monitor~~ing system~~ inoperable ~~with only the containment gaseous radioactivity monitor operable~~, perform ~~an RCS water inventory balance~~ at least once per 24* hours and analyze grab samples of the containment atmosphere at least once per 12 hours, and either restore the sump inlet flow monitor~~ing system~~ to OPERABLE status within 7 days or restore the containment particulate radioactivity monitor to OPERABLE status within 7 days ~~and enter action a. above with the time in this action applied against the allowed outage time of action a.;~~ ~~otherwise,~~ be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

SR 3.4.13.1 A02

A03

Required Action A.1

Required Action C.1

Required Action C.2

Condition C Note

ACTION D

MODE 3

MODE 5

36

Condition B ~~c.~~ With the required radioactivity monitor inoperable, analyze grab samples of the containment atmosphere or perform ~~a RCS water inventory balance~~ at least once per 24* hours, and restore the required radioactivity monitor to OPERABLE status within 30 days; ~~otherwise,~~ be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

SR 3.4.13.1 A02

Required Action B.1.1

Required Action B.1.2

Required Action B.2

ACTION D

MODE 3

MODE 5

36

ACTION E ~~d.~~ ~~With~~ all required monitors inoperable, enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

~~4.4.6.1~~ ~~The RCS leakage detection instruments shall be demonstrated OPERABLE by:~~

SR 3.4.15.1, SR 3.4.15.2 SR 3.4.15.4 ~~a.~~ Performa~~nce of the~~ CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor in accordance with ~~surveillance 4.3.3.1.~~ the Surveillance Frequency Control Program.

A02

SR 3.4.15.3 b. Performa~~nce of the~~ CHANNEL CALIBRATION of the reactor cavity sump inlet flow monitor~~ing system~~ in accordance with the Surveillance Frequency Control Program.

Required Action A.1 Note, B.1.2 Note, * Not required to be performed until 12 hours after establishment of steady state operation.

~~INSTRUMENTATION~~ ← REACTOR COOLANT SYSTEMS (RCS)

RCS Leakage Detection

3/4.3.3 ~~MONITORING~~ INSTRUMENTATION

A04

~~RADIATION MONITORING INSTRUMENTATION~~

LIMITING CONDITION FOR OPERATION

(gaseous or particulate)

One containment

LCO 3.4.15.b 3.3.3.1 ~~The~~ radiation monitor~~ing instrumentation~~ channel~~s~~ ~~shown in Table 3.3-6~~ shall be OPERABLE ~~with their alarm/trip setpoints within the specified limits~~.

Applicability **APPLICABILITY**: ~~As shown in Table 3.3-6.~~ ← MODES 1, 2, 3, and 4

See ITS 3.3.6
ITS 3.3.7
ITS 3.3.8
ITS 3.3.9

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

required

ACTIONS b. With one or more ~~radiation monitoring~~ channel~~s~~ inoperable, take the ACTION shown in ~~Table 3.3-6~~. ← Specification 3.4.15

A04

~~c. The provisions of Specification 3.0.3 are not applicable.~~

A05

SURVEILLANCE REQUIREMENTS

required containment

SR 3.4.15.1
SR 3.4.15.2
SR 3.4.15.4 4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program.

A04

4.3.3.2 In accordance with the Surveillance Frequency Control Program, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

See ITS 3.3.7

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION	
~~1. AREA MONITORS~~						
a. Fuel Storage Pool Area						
i. Criticality and Ventilation System Isolation Monitor	4	*	≤ 20 mR/hr	10^{-1} – 10^{4} mR/hr	22	See ITS 3.3.8
b. Containment Isolation	3	****	≤ 90 mR/hr	1 – 10^{7} mR/hr	25	See ITS 3.3.6
c. Containment Area – Hi Range	1	1, 2, 3 & 4	Not Applicable	1 - 10^{7} R/hr	27	See ITS 3.3.9
d. Control Room Isolation	1 per intake	ALL MODES	≤ 320 cpm	10^{-7}– 10^{-2} µCi/cc	26	See ITS 3.3.7
2. PROCESS MONITORS						
LCO 3.4.15 a. Containment						
i. Gaseous Activity RCS Leakage Detection	1 (A06)	1, 2, 3 & 4	~~Not Applicable~~	~~10^{-7} – 10^{-2} µCi/cc~~ (LA01)	23	
ii. Particulate Activity RCS Leakage Detection	1 (or)	1, 2, 3 & 4	~~Not Applicable~~	~~10 – 10^{7} cpm~~	23	

* With fuel in the storage pool or building. (See ITS 3.3.8)

**** During movement of recently irradiated fuel assemblies within containment. (See ITS 3.3.6)

~~TABLE 3.3-6 (Continued)~~

~~ACTION STATEMENTS~~

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTIONS

~~ACTION 23~~ - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification ~~3.4.6.1~~. 3.4.15

A06

ACTION 24 - DELETED

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

See ITS 3.3.9

DISCUSSION OF CHANGES
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.6.1 Actions a, b, and c, state, in part, perform a RCS water inventory balance at least once per 24 hours. Unit 1 CTS 4.4.6.2.c and Unit 2 CTS 4.4.6.2.1c (i.e., RCS Operational Leakage) require demonstration that the RCS operational LEAKAGE is within limits by performance of a Reactor Coolant System water inventory balance. ITS 3.4.15 states “Perform SR 3.4.13.1 once per 24 hours.” ITS SR 3.4.13.1 requires verification that the RCS operational LEAKAGE is within limits by performance of an RCS water inventory balance.” This changes the CTS by explicitly requiring that the RCS water balance Surveillance be performed.

CTS 4.4.6.1.a requires performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor in accordance with Surveillance Requirement 4.3.3.1. The CTS 4.3.3.1 Surveillance Frequency for the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST of each radiation monitoring instrumentation channel is in accordance with the Surveillance Frequency Control Program. Therefore, the Surveillance Frequency for CTS 4.4.6.1.a is changed to “in accordance with the Surveillance Frequency Control Program” consistent with the referenced CTS 4.3.3.1 Surveillance Requirement. ITS SR 3.4.15.1, SR 3.4.15,2, and SR 3.4.15.4 Frequencies are annotated “in accordance with the Surveillance Frequency Control Program” consistent with CTS 4.4.3.1.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A03 CTS 3.4.6.1 Action b states, in part, with the reactor cavity sump inlet flow monitoring system inoperable with only the containment gaseous radioactivity monitor operable, perform an RCS water inventory balance at least once per 24 hours and analyze grab samples of the containment atmosphere at least once per 12 hours, and either restore the containment particulate radioactivity monitor to OPERABLE status within 7 days, and enter action a. above with the time in this action applied against the allowed outage time of action a.

ITS 3.4.15 ACTION C does not include the requirement “perform an RCS water inventory balance at least once per 24 hours” since with the reactor cavity sump inlet flow monitoring system inoperable, ACTION A is also entered (LCO 3.0.2), and Required Action A.1 performs the RCS water inventory balance using SR 3.4.13.1.

ITS 3.4.15 ACTION C does not include the requirement to "restore the containment particulate radioactivity monitor to OPERABLE status within 7 days." ITS Required Action C.2 restores the reactor cavity sump inlet flow monitor to OPERABLE status within 7 days. The Condition C note implies that the containment particulate radioactivity monitor is inoperable, and though not a Required Action, it remains an option to restore the containment particulate radioactivity monitor to OPERABLE status. Required Action C.2 allows 7 days to restore the reactor cavity sump inlet flow monitor to OPERABLE status before entering Condition D; therefore, the Completion Time of 7 days is also allowed to restore the containment particulate radioactivity monitor to OPERABLE status.

ITS 3.4.15 ACTION C does not include an action similar to the CTS Action b., which requires entry into Action a. and applying the time in Action b. to the allowed outage time of Action a. This is not necessary since with the reactor cavity sump inlet flow monitoring system inoperable, ACTION A is also entered (LCO 3.0.2), therefore, the allowed outage time is also accrued for ACTION A.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A04 CTS 3.3.3.1 specifies requirements for radiation monitoring instrumentation and list the requirements for each instrument in CTS Table 3.3-6, including the containment gaseous and particulate activity RCS leakage detection instrumentation. ITS 3.4.15 provides, in part, requirements for the containment gaseous and particulate activity RCS leakage detection instrumentation consistent with the requirements of CTS 3.4.6.1 and provides the requirements in the LCO, Applicability, ACTIONS, and Surveillance Requirements. This changes the CTS by presenting the gaseous and particulate activity RCS leakage detection instrument requirements without use of an instrument table. This change is acceptable as it results solely from the change in the format and presentation of the CTS necessary to conform to the ISTS. As the proposed change is the result of changes in the format and presentation of the CTS requirements, it is designated administrative.

A05 CTS 3.3.3.1 Action c states that the provisions of Specification 3.0.3 are not applicable. ITS 3.4.15 does not include an allowance waving the provisions of LCO 3.0.3. This changes the CTS by eliminating an instrumentation action requirement that conflicts with an action in Specification 3.4.6.1.

The purpose of CTS 3.3.3.1 Action c is to preclude action be taken per Specification 3.0.3 (ITS LCO 3.0.3) to shutdown the unit in the event there are no actions to perform, or the actions cannot be performed within the required time when radiation monitoring instrumentation is inoperable. However, a portion of the radiation monitoring instrumentation is also utilized for RCS leakage detection. CTS 3.3.3.1 Action c states that the provisions of Specification 3.0.3 are not applicable. Yet, CTS Table 3.3-6 Action 14 (Unit 1) and Action 23 (Unit 2) require complying with Specification 3.4.6.1. CTS 3.4.6.1 Action d requires entry into Specification 3.0.3 when all required leakage detection monitors, including the containment gaseous and particulate activity RCS leakage detection instrumentation, are inoperable. Therefore, the requirements of CTS 3.3.3.1 and

CTS 3.4.6.1 appear to conflict with regard to CTS 3.0.3 compliance. This change eliminates conflicting statements regarding LCO 3.0.3 compliance by removing the Specification 3.0.3 waiver for the containment gaseous and particulate activity RCS leakage detection instrumentation and preserves the intent of the CTS requirement to apply Specification 3.0.3 when all required leakage detection monitors, including the containment gaseous and particulate activity RCS leakage detection instruments, are inoperable. This change is designated as administrative and is acceptable because it does not result in a technical change to the CTS.

A06 CTS 3.3.3.1 Table 3.3-6 requires a minimum of one channel to be OPERABLE for the containment gaseous activity RCS leakage detection monitor (Instrument 2.a.i) and the containment particulate activity RCS leakage detection monitor (Instrument 2.a.ii). The associated CTS Table 3.3-6 Action requires action to be taken pursuant to Specification 3.4.6.1. CTS 3.4.6.1.b requires either one gaseous channel or one particulate channel of the containment atmosphere radioactivity monitor to be OPERABLE and provides actions when one "required" containment atmosphere radioactivity monitor is inoperable. Consistent with CTS 3.4.6.1, ITS LCO 3.4.15.b also requires one channel of the containment atmosphere radioactivity monitor (gaseous or particulate) to be OPERABLE and ACTIONS when one "required" channel is inoperable. This changes the CTS presentation by eliminating potential confusion of the existing requirements by clarifying the intent that one channel of either the gaseous or particulate activity RCS leakage detection instrument must be OPERABLE. This change is acceptable as it represents a change in the format and presentation of the CTS to more accurately reflect the intent of the existing requirements as specified in CTS 3.4.6.1. As the proposed change is the result of changes in the format and presentation of the CTS requirements, it is designated administrative.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 1 - Removing Details of System Design and System Description, Including Design Limits*) CTS 3.3.3.1 Table 3.3-6 for radiation monitoring instrumentation has a column specifying the Measurement Range and a specific measurement range for the containment gaseous and particulate activity RCS leakage detection instrumentation. ITS 3.4.15 does not retain the measurement range of the containment gaseous and particulate activity RCS leakage detection instrumentation. This changes the CTS by moving the instrument measurement range to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the number of required channels and the appropriate Condition to enter if a required channel becomes inoperable. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

3.4.6.1 LCO 3.4.15 ~~[Two of]~~ the following RCS leakage detection instrumentation shall be OPERABLE: 2

a. One ~~containment~~ (reactor cavity) sump (inlet flow) monitor, (and) 1

b. One containment atmosphere radioactivity monitor (gaseous or particulate)~~, and~~

~~[c. One containment air cooler condensate flow rate monitor.]~~ 2

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a. A. (Reactor cavity) ~~Containment~~ sump (inlet flow) monitor inoperable.	A.1 ----------NOTE---------- Not required until 12 hours after establishment of steady state operation. ---------- Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	A.2 Restore (reactor cavity) ~~containment~~ sump (inlet flow) monitor to OPERABLE status.	30 days

1

1

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action c. B. Required containment atmosphere radioactivity monitor inoperable.	B.1.1 Analyze grab samples of the containment atmosphere. OR B.1.2 ----------NOTE---------- Not required until 12 hours after establishment of steady state operation. ---------- Perform SR 3.4.13.1. AND B.2~~.1~~ Restore required containment atmosphere radioactivity monitor to OPERABLE status. ~~OR~~ ~~[B.2.2 Verify containment air cooler condensate flow rate monitor is OPERABLE.~~	Once per 24 hours Once per 24 hours 30 days ~~30 days]~~
~~C. [Containment air cooler condensate flow rate monitor inoperable.~~	~~C.1 Perform SR 3.4.15.1.~~ ~~OR~~ ~~C.2 ----------NOTE---------- Not required until 12 hours after establishment of steady state operation. ---------- Perform SR 3.4.13.1.~~	~~Once per 8 hours~~ ~~Once per 24 hours]~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action b. ----------NOTE---------- Only applicable when the containment atmosphere gaseous radiation monitor is the only OPERABLE monitor. ---------- Reactor cavity inlet flow D. ~~Containment~~t sump monitor inoperable. (C) AND ~~[Containment air cooler condensate flow rate monitor inoperable.]~~	~~D~~.1 Analyze grab samples of the containment atmosphere. (C) AND ~~D~~.2~~.1~~ Restore ~~containment~~ reactor cavity sump inlet flow monitor to OPERABLE status. (C) OR ~~[D.2.2 Restore containment air cooler condensate flow rate monitor to OPERABLE status.]~~	Once per 12 hours 7 days ~~7 days~~	1 1 1 2
~~E. [Required containment atmosphere radioactivity monitor inoperable.~~ AND ~~[Containment air cooler condensate flow rate monitor inoperable.]~~	~~E.1 Restore required containment atmosphere radioactivity monitor to OPERABLE status.~~ OR ~~E.2 Restore containment air cooler condensate flow rate monitor to OPERABLE status.~~	~~30 days~~ ~~30 days]~~	2 2
Action a., b., c. F. Required Action and associated Completion Time not met. (D)	~~F~~.1 Be in MODE 3. (D) AND ~~F~~.2 Be in MODE 5. (D)	6 hours 36 hours	1 1
Action d. ~~G~~. All required monitors inoperable. (E)	~~G~~.1 Enter LCO 3.0.3. (E)	Immediately	1

ACTIONS (continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.6.1	SR 3.4.15.1	Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor.	~~[[12] hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.6.1	SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitor.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.6.2	SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required ~~containment~~ sump monitor. reactor cavity inlet flow	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.6.1	SR 3.4.15.4	Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
	~~SR 3.4.15.5~~	~~[Perform CHANNEL CALIBRATION of the required containment air cooler condensate flow rate monitor.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~	2 2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

3.4.6.1 LCO 3.4.15 ~~[Two of]~~ the following RCS leakage detection instrumentation shall be OPERABLE: 2

reactor cavity inlet flow

a. One ~~containment~~ sump monitor, and 1

b. One containment atmosphere radioactivity monitor (gaseous or particulate)~~, and~~

~~[c. One containment air cooler condensate flow rate monitor.]~~ 2

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a. Reactor cavity inlet flow A. ~~Containment~~ sump monitor inoperable.	A.1 ----------NOTE---------- Not required until 12 hours after establishment of steady state operation. ---------- Perform SR 3.4.13.1.	1 Once per 24 hours
	AND	
	A.2 Restore ~~containment~~ reactor cavity sump inlet flow monitor to OPERABLE status.	1 30 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action c. B. Required containment atmosphere radioactivity monitor inoperable.	B.1.1 Analyze grab samples of the containment atmosphere.	Once per 24 hours	
	OR		
	B.1.2 ----------NOTE---------- Not required until 12 hours after establishment of steady state operation. ---------- Perform SR 3.4.13.1.	Once per 24 hours	
	AND		
	B.2~~.1~~ Restore required containment atmosphere radioactivity monitor to OPERABLE status.	30 days	1
	~~OR~~		
	~~[B.2.2 Verify containment air cooler condensate flow rate monitor is OPERABLE.~~	~~30 days]~~	2
~~C. [Containment air cooler condensate flow rate monitor inoperable.~~	~~C.1 Perform SR 3.4.15.1.~~	~~Once per 8 hours~~	
	~~OR~~		
	~~C.2 ----------NOTE---------- Not required until 12 hours after establishment of steady state operation. ---------- Perform SR 3.4.13.1.~~	~~Once per 24 hours]~~	2

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action b.	----------NOTE---------- Only applicable when the containment atmosphere gaseous radiation monitor is the only OPERABLE monitor. ----------	~~D~~ C.1 Analyze grab samples of the containment atmosphere.	Once per 12 hours	1
		AND		1
	~~D~~ C. ~~Containment~~ Reactor cavity sump inlet flow monitor inoperable.	~~D~~ C.2~~.1~~ Restore ~~containment~~ reactor cavity sump inlet flow monitor to OPERABLE status.	7 days	1
		OR		
	~~AND~~ ~~[Containment air cooler condensate flow rate monitor inoperable.]~~	~~[D.2.2~~ ~~Restore containment air cooler condensate flow rate monitor to OPERABLE status.]~~	~~7 days~~	2
	~~E.~~ ~~[Required containment atmosphere radioactivity monitor inoperable.~~	~~E.1~~ ~~Restore required containment atmosphere radioactivity monitor to OPERABLE status.~~	~~30 days~~	2
	~~AND~~	~~OR~~		
	~~[Containment air cooler condensate flow rate monitor inoperable.]~~	~~E.2~~ ~~Restore containment air cooler condensate flow rate monitor to OPERABLE status.~~	~~30 days]~~	2
Action a., b., c.	~~F~~ D. Required Action and associated Completion Time not met.	~~F~~ D.1 Be in MODE 3.	6 hours	1
		AND		
		~~F~~ D.2 Be in MODE 5.	36 hours	1
Action d.	~~G~~ E. All required monitors inoperable.	~~G~~ E.1 Enter LCO 3.0.3.	Immediately	1

ACTIONS (continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.6.1	SR 3.4.15.1	Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor.	~~[[12] hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.6.1	SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitor.	~~[92 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.4.6.2	SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required ~~containment~~ sump monitor. reactor cavity inlet flow	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.6.1	SR 3.4.15.4	Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
	~~SR 3.4.15.5~~	~~[Perform CHANNEL CALIBRATION of the required containment air cooler condensate flow rate monitor.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~	2 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45, Revision 0, (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. ~~[In addition to meeting the OPERABILITY requirements, the monitors are typically set to provide the most sensitive response without causing an excessive number of spurious alarms.]~~ 2

reactor cavity

The ~~containment~~ sump used to collect unidentified LEAKAGE ~~[~~is~~]~~ ~~[(or) and the containment air cooler condensate flow rate monitor] [are]~~ instrumented to alarm for increases above in the normal flow rates. 1 2

The reactor coolant contains radioactivity that, when released to the containment, may be detected by radiation monitoring instrumentation. Radioactivity detection systems are included for monitoring both particulate and gaseous activities, because of their sensitivities and rapid responses to RCS LEAKAGE. 1

The measurement range of the containment particulate and gaseous radioactivity monitoring instruments is 10 to 10^6 cpm.

~~Other indications may be used to detect an increase in unidentified LEAKAGE; however, they are not required to be OPERABLE by this LCO. An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE.~~ 1

~~Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump [and condensate flow from air coolers]. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.~~ 1

BASES

BACKGROUND (continued)

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS LEAKAGE into the containment. The relevance of temperature and pressure measurements is affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

The above-mentioned LEAKAGE detection methods or systems differ in sensitivity and response time. ~~[Some of these systems could serve as early alarm systems signaling the operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required.]~~ 2

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area are necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should leakage occur detrimental to the safety of the facility and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide confidence that small amounts of unidentified LEAKAGE are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

two

The LCO requires ~~[three]~~ instruments to be OPERABLE. 2

reactor cavity

The ~~containment~~ sump is used to collect unidentified LEAKAGE. ~~[The containment sump consists of the normal sump and the emergency sump.~~ The LCO requirements apply to the total amount of unidentified LEAKAGE collected in ~~[~~the~~][both]~~ sump~~[s].]~~ The monitor on the ~~containment~~ sump detects ~~[level or~~ flow rate ~~or the operating frequency of a pump]~~ and is instrumented to detect when there is ~~[leakage of] [~~an 1 2

reactor cavity

BASES

LCO (continued)

increase above the normal value by] 1 gpm. The identification of [an increase in] unidentified LEAKAGE will be delayed by the time required for the unidentified LEAKAGE to travel to the containment sump and it may take longer than one hour to detect a 1 gpm increase in unidentified LEAKAGE, depending on the origin and magnitude of the LEAKAGE. This sensitivity is acceptable for containment sump monitor OPERABILITY.

reactor cavity

reactor cavity

inlet flow

1

1

The reactor coolant contains radioactivity that, when released to the containment, can be detected by the gaseous or particulate containment atmosphere radioactivity monitor. Only one of the two detectors is required to be OPERABLE. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE, but have recognized limitations. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. If there are few fuel element cladding defects and low levels of activation products, it may not be possible for the gaseous or particulate containment atmosphere radioactivity monitors to detect a 1 gpm increase within 1 hour during normal operation. However, the gaseous or particulate containment atmosphere radioactivity monitor is OPERABLE when it is capable of detecting a 1 gpm increase in unidentified LEAKAGE within 1 hour given an RCS activity equivalent to that assumed in the design calculations for the monitors (Reference 3).

[An increase in humidity of the containment atmosphere could indicate the release of water vapor to the containment. Condensate flow from air coolers is instrumented to detect when there is an increase above the normal value by 1 gpm. The time required to detect a 1 gpm increase above the normal value varies based on environmental and system conditions and may take longer than 1 hour. This sensitivity is acceptable for containment air cooler condensate flow rate monitor OPERABILITY.]

2

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor, in combination with a particulate or gaseous radioactivity monitor [and a containment air cooler condensate flow rate monitor], provides an acceptable minimum.

inlet flow

reactor cavity

1

2

APPLICABILITY Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

1

BASES

APPLICABILITY (continued)

In MODE 5 or 6, the temperature is ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

reactor cavity

inlet flow

If the ~~containment~~ sump monitor is inoperable, no other form of sampling can provide the equivalent information. 1

However, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the containment atmosphere radioactivity monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown~~, and [RCP seal injection and return flows]~~). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. 2

Restoration of the sump monitor to OPERABLE status is required to regain the function in a Completion Time of 30 days after the monitor's failure. This time is acceptable considering the frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

B.1.1, B.1.2, ~~B.2.1,~~ and B.2~~.2~~ 1

With both gaseous and particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed, or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. With a sample obtained and analyzed or an inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors.

~~[Alternatively, continued operation is allowed if the air cooler condensate flow rate monitoring system is OPERABLE, provided grab samples are taken or water inventory balance performed every 24 hours.]~~ 2

BASES

ACTIONS (continued)

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown~~, and [RCP seal injection and return flows]~~). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leakage detection is available. 2

~~[C.1 and C.2~~

~~If the containment air cooler condensate flow rate monitor is inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed, or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or an inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment air cooler condensate flow rate monitor to OPERABLE status.~~ 2

~~The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and [RCP seal injection and return flows]). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.]~~ 2

C.1 and C.2

~~D.1, D.2.1, and D.2.2~~ 1

reactor cavity

inlet flow

With the ~~containment~~ sump monitor~~, [and the containment air cooler condensate flow rate monitor]~~ inoperable, the only means of detecting LEAKAGE is the required containment atmosphere radiation monitor. A Note clarifies that this Condition is only applicable when the only OPERABLE monitor is the required containment atmosphere gaseous radiation monitor. The containment atmosphere gaseous radioactivity monitor typically cannot detect a 1 gpm leak within one hour when RCS activity is low. In addition, this configuration does not provide the required diverse means of leakage detection. Indirect methods of monitoring RCS leakage must be implemented. Grab samples of the containment atmosphere must be taken and analyzed every 12 hours to provide 2

BASES

ACTIONS (continued)

alternate periodic information. The 12 hour interval is sufficient to detect increasing RCS leakage. The Required Action provides 7 days to restore another RCS leakage monitor to OPERABLE status to regain the intended leakage detection diversity. The 7 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

The Note assumes the containment particulate radioactivity monitor is inoperable. If the containment particulate radioactivity monitor is restored to OPERABLE status within the 7 day Completion Time to restore the reactor cavity sump inlet flow monitor to OPERABLE status, the Condition may be exited.

1

~~E.1 and E.2~~

~~If the required containment atmosphere radioactivity monitor [and the containment air cooler condensate flow rate monitor] are inoperable, the only means of detecting leakage is the containment sump monitor. This Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Times ensure that the plant will not be operated in a reduced configuration for a lengthy time period.~~

1

D D

~~F~~.1 and ~~F~~.2

1

or

If any Required Action of Condition A, B, ~~[~~C~~], [D] or [E]~~ cannot be met within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

1

E

~~G~~.1

1

If all required monitors are inoperable, no automatic means of monitoring leakage are available and immediate plant shutdown in accordance with LCO 3.0.3 is required.

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence the channel is operating properly. ~~[The Frequency of [12] hours is based on instrument reliability and is reasonable for detecting off normal conditions.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitors. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other

BASES

SURVEILLANCE REQUIREMENTS (continued)

Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. ~~[The Frequency of 92 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~]~~

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

and

SR 3.4.15.3~~,~~ SR 3.4.15.4~~, and [SR 3.4.15.5]~~ 1

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. [~~ The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Operating experience has shown this Frequency is acceptable.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

BASES

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.

2. Regulatory Guide 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3. FSAR, Section [ ].

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5.2.4

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45, Revision 0, (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. ~~[In addition to meeting the OPERABILITY requirements, the monitors are typically set to provide the most sensitive response without causing an excessive number of spurious alarms.]~~ 2

reactor cavity

The ~~containment~~ sump used to collect unidentified LEAKAGE ~~[~~is~~]~~ ~~[(or) and the containment air cooler condensate flow rate monitor] [are]~~ instrumented to alarm for increases above in the normal flow rates. 1 2

The reactor coolant contains radioactivity that, when released to the containment, may be detected by radiation monitoring instrumentation. Radioactivity detection systems are included for monitoring both particulate and gaseous activities, because of their sensitivities and rapid responses to RCS LEAKAGE. 1

The measurement range of the containment particulate radioactivity monitoring instrument is 10 to 10^7 cpm. The measurement range of the containment gaseous radioactivity monitoring instrument is 10^{-7} to 10^{-2} µCi/cc.

~~Other indications may be used to detect an increase in unidentified LEAKAGE; however, they are not required to be OPERABLE by this LCO. An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE.~~ 1

~~Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump [and condensate flow from air coolers]. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.~~ 1

BASES

BACKGROUND (continued)

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS LEAKAGE into the containment. The relevance of temperature and pressure measurements is affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

The above-mentioned LEAKAGE detection methods or systems differ in sensitivity and response time. ~~[Some of these systems could serve as early alarm systems signaling the operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required.]~~ 2

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area are necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should leakage occur detrimental to the safety of the facility and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide confidence that small amounts of unidentified LEAKAGE are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

two

The LCO requires ~~[three]~~ instruments to be OPERABLE. 2

reactor cavity

The ~~containment~~ sump is used to collect unidentified LEAKAGE. ~~[The containment sump consists of the normal sump and the emergency sump.~~ The LCO requirements apply to the total amount of unidentified LEAKAGE collected in ~~[~~the~~][both]~~ sump~~[s].]~~ The monitor on the ~~containment~~ sump detects ~~[level or~~ flow rate ~~or the operating frequency of a pump]~~ and is instrumented to detect when there is ~~[leakage of]~~ [an 1 2

reactor cavity

BASES

LCO (continued)

increase above the normal value by] 1 gpm. The identification of [an increase in] unidentified LEAKAGE will be delayed by the time required for the unidentified LEAKAGE to travel to the containment sump and it may take longer than one hour to detect a 1 gpm increase in unidentified LEAKAGE, depending on the origin and magnitude of the LEAKAGE. This sensitivity is acceptable for containment sump monitor OPERABILITY.

reactor cavity

reactor cavity

inlet flow

1

1

The reactor coolant contains radioactivity that, when released to the containment, can be detected by the gaseous or particulate containment atmosphere radioactivity monitor. Only one of the two detectors is required to be OPERABLE. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE, but have recognized limitations. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. If there are few fuel element cladding defects and low levels of activation products, it may not be possible for the gaseous or particulate containment atmosphere radioactivity monitors to detect a 1 gpm increase within 1 hour during normal operation. However, the gaseous or particulate containment atmosphere radioactivity monitor is OPERABLE when it is capable of detecting a 1 gpm increase in unidentified LEAKAGE within 1 hour given an RCS activity equivalent to that assumed in the design calculations for the monitors (Reference 3).

[An increase in humidity of the containment atmosphere could indicate the release of water vapor to the containment. Condensate flow from air coolers is instrumented to detect when there is an increase above the normal value by 1 gpm. The time required to detect a 1 gpm increase above the normal value varies based on environmental and system conditions and may take longer than 1 hour. This sensitivity is acceptable for containment air cooler condensate flow rate monitor OPERABILITY.]

2

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor, in combination with a particulate or gaseous radioactivity monitor [and a containment air cooler condensate flow rate monitor], provides an acceptable minimum.

inlet flow

reactor cavity

1

2

APPLICABILITY Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

BASES

APPLICABILITY (continued)

In MODE 5 or 6, the temperature is ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

reactor cavity

inlet flow

If the ~~containment~~ sump monitor is inoperable, no other form of sampling can provide the equivalent information. 1

However, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. Together with the containment atmosphere radioactivity monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown~~, and [RCP seal injection and return flows]~~). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. 2

Restoration of the sump monitor to OPERABLE status is required to regain the function in a Completion Time of 30 days after the monitor's failure. This time is acceptable considering the frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

B.1.1, B.1.2, ~~B.2.1,~~ and B.2~~.2~~ 1

With both gaseous and particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed, or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. With a sample obtained and analyzed or an inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors.

~~[Alternatively, continued operation is allowed if the air cooler condensate flow rate monitoring system is OPERABLE, provided grab samples are taken or water inventory balance performed every 24 hours.]~~ 2

BASES

ACTIONS (continued)

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown~~, and [RCP seal injection and return flows]~~). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leakage detection is available. 2

~~[C.1 and C.2~~

~~If the containment air cooler condensate flow rate monitor is inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed, or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or an inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment air cooler condensate flow rate monitor to OPERABLE status.~~ 2

~~The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and [RCP seal injection and return flows]). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.]~~ 2

C.1 and C.2

~~D.1, D.2.1, and D.2.2~~ 1

reactor cavity

inlet flow

With the ~~containment~~ sump monitor~~, [and the containment air cooler condensate flow rate monitor]~~ inoperable, the only means of detecting LEAKAGE is the required containment atmosphere radiation monitor. A Note clarifies that this Condition is only applicable when the only OPERABLE monitor is the required containment atmosphere gaseous radiation monitor. The containment atmosphere gaseous radioactivity monitor typically cannot detect a 1 gpm leak within one hour when RCS activity is low. In addition, this configuration does not provide the required diverse means of leakage detection. Indirect methods of monitoring RCS leakage must be implemented. Grab samples of the containment atmosphere must be taken and analyzed every 12 hours to provide 2

BASES

ACTIONS (continued)

alternate periodic information. The 12 hour interval is sufficient to detect increasing RCS leakage. The Required Action provides 7 days to restore another RCS leakage monitor to OPERABLE status to regain the intended leakage detection diversity. The 7 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

The Note assumes the containment particulate radioactivity monitor is inoperable. If the containment particulate radioactivity monitor is restored to OPERABLE status within the 7 day Completion Time to restore the reactor cavity sump inlet flow monitor to OPERABLE status, the Condition may be exited.

1

~~E.1 and E.2~~

~~If the required containment atmosphere radioactivity monitor [and the containment air cooler condensate flow rate monitor] are inoperable, the only means of detecting leakage is the containment sump monitor. This Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Times ensure that the plant will not be operated in a reduced configuration for a lengthy time period.~~

1

D D

~~F~~.1 and ~~F~~.2

1

or

If any Required Action of Condition A, B, ~~[~~C~~], [D] or [E]~~ cannot be met within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

1

E

~~G~~.1

1

If all required monitors are inoperable, no automatic means of monitoring leakage are available and immediate plant shutdown in accordance with LCO 3.0.3 is required.

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence the channel is operating properly. ~~[The Frequency of [12] hours is based on instrument reliability and is reasonable for detecting off normal conditions.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitors. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other

BASES

SURVEILLANCE REQUIREMENTS (continued)

Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. ~~[The Frequency of 92 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~]~~

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

and

SR 3.4.15.3~~,~~ SR 3.4.15.4~~, and [SR 3.4.15.5]~~ 1

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. [~~ The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Operating experience has shown this Frequency is acceptable.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

BASES

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.

2. Regulatory Guide 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3. FSAR, Section [ ].

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5.2.5

1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.15, BASES, RCS LEAKAGE DETECTION INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.15, RCS LEAKAGE DETECTION INSTRUMENTATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 16

3.4.16, RCS Specific Activity

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.16 ~~3.4.8~~ ~~The~~ specific activity ~~of the primary coolant~~ shall be ~~limited~~ ~~to:~~

RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133

within limits.

SR 3.4.16.2 ~~a.~~ ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

SR 3.4.16.1 ~~b.~~ ≤ 518.9 μCi/gram DOSE EQUIVALENT XE-133.

Applicability **APPLICABILITY:** MODES 1, 2, 3, and 4.

ACTION:

Condition A
Required Action A.1

~~a.~~ ~~With the specific activity of the primary coolant >1.0 μCi/gram~~ DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is ≤ 60.0 μCi/gram once per four hours.

not within limits

~~b.~~ With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131, but ≤ 60.0 μCi/gram DOSE EQUIVALENT I-131, operation may continue for up to 48 hours ~~while efforts are made to~~ restore DOSE EQUIVALENT I-131 to within ~~the 1.0 μCi/gram~~ limit.

Required Action A.2

LCO 3.0.4.c is applicable.

Required Action A.1 and A.2 Note

Condition C

~~c.~~ With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for greater than 48 hours during one continuous time interval, or > 60.0 μCi/gram DOSE EQUIVALENT I-131,

Required Action C.1 and C.2

be in ~~HOT STANDBY~~ within 6 hours and ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

MODE 3

MODE 5

36

~~d.~~ With the specific activity of the primary coolant > 518.9 μCi/gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours ~~while efforts are made to~~ restore DOSE EQUIVALENT XE-133 to within ~~the 518.9 μCi/gram DOSE EQUIVALENT XE-133~~ limit.

Required Action B.1

LCO 3.0.4.c is applicable.

Required Action B.1 Note

Condition C

~~e.~~ With the specific activity of the primary coolant > 518.9 μCi/gram DOSE EQUIVALENT XE-133 for greater than 48 hours during one continuous time interval,

Required Action C.1 and C.2

be in ~~HOT STANDBY~~ within 6 hours and ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

MODE 3

MODE 5

36

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1
SR 3.4.16.2

~~4.4.8~~ ~~The~~ specific activity ~~of the primary coolant shall be determined to be~~ within ~~the~~ limits ~~by performing sampling and analysis as described in Table 4.4-4~~.

Verify reactor coolant DOSE EQUIVALENT XE-133; Verify reactor coolant DOSE EQUIVALENT I-131

A01

REACTOR COOLANT SYSTEM

~~DELETED~~

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS

	TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
SR 3.4.16.1	1. DOSE EQUIVALENT XE-133 Determination	SFCP	1, 2, 3 and 4
SR 3.4.16.2	2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	SFCP	1
Required Action A.1	3. Isotopic Analysis for Iodine Including I-131, I-132, I-133, I-134, and I-135	a) Once per 4 hours, whenever the DOSE EQUIVALENT I-131 exceeds 1.0 μCi/gram, and	1#, 2#, 3#, and 4#
SR 3.4.16.2 Frequency		b) One sample between 2 and 6 hours following a THERMAL POWER change ≥ 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

LA01 A02 after A03

Until the specific activity of the primary coolant system is restored within its limits. A02

A01

~~DELETED~~

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.16

~~3.4.8~~ ~~The~~ specific activity ~~of the primary coolant~~ shall be ~~limited~~ ~~to:~~

SR 3.4.16.2

~~a.~~ Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

SR 3.4.16.1

~~b.~~ Less than or equal to 518.9 microcuries/gram DOSE EQUIVALENT XE-133.

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

Condition A
Required Action A.1

~~a.~~ ~~With the specific activity of the primary coolant > 1.0 µCi/gram~~ DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is ≤ 60.0 µCi/gram once per four hours.

Required Action A.2

~~b.~~ With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131, but ≤ 60.0 µCi/gram DOSE EQUIVALENT I-131, operation may continue for up to 48 hours ~~while efforts are made to~~ restore DOSE EQUIVALENT I-131 to within ~~the 1.0 µCi/gram~~ limit.

Required Action A.1 and A.2 Note

LCO 3.0.4.c is applicable.

Condition C
Required Action C.1 and C.2

~~c.~~ With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 for greater than 48 hours during one continuous time interval, or > 60.0 µCi/gram DOSE EQUIVALENT I-131, be in ~~HOT STANDBY~~ within 6 hours and ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

Required Action B.1

~~d.~~ With the specific activity of the primary coolant > 518.9 µCi/gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours ~~while efforts are made to~~ restore DOSE EQUIVALENT XE-133 to within ~~the 518.9 µCi/gram DOSE EQUIVALENT XE-133~~ limit.

Required Action B.1 Note

LCO 3.0.4.c is applicable.

Condition C
Required Action C.1 and C.2

~~e.~~ With the specific activity of the primary coolant > 518.9 µCi/gram DOSE EQUIVALENT XE-133 for greater than 48 hours during one continuous time interval, be in ~~HOT STANDBY~~ within 6 hours and ~~COLD SHUTDOWN~~ within the ~~following~~ ~~30~~ hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1
SR 3.4.16.2

~~4.4.8~~ ~~The~~ specific activity ~~of the primary coolant shall be determined to be~~ within ~~the~~ limits ~~by performing sampling and analysis as described in Table 4.4-4~~.

A01

~~REACTOR COOLANT SYSTEM~~

~~DELETED~~

A01

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS

	TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
SR 3.4.16.1	1. DOSE EQUIVALENT XE-133 Determination	SFCP	1, 2, 3, and 4
SR 3.4.16.2	2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	SFCP	1
Required Action A.1	3. Isotopic Analysis for Iodine Including I-131, I-132, I-133, I-134, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 micro-Ci/gram, DOSE EQUIVALENT I-131, and	1#, 2#, 3#, and 4#
SR 3.4.16.2 Frequency		b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

LA01

A02

after

≥

A03

Until the specific activity of the primary coolant system is restored within its limits.

A02

A01

~~DELETED~~

DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.4.8 (MODES 1, 2, 3, and 4), ACTION a, requires Table 4.4-4, Sampling Test 3.a, Isotopic Analysis for Iodine, to be performed every 4 hours until the specific activity of the primary coolant is restored to within its limits. This is also restated in CTS Table 4.4-4, Footnote #. ITS 3.4.16 Required Action A.1 essentially requires the same analysis; however, the explicit statement to perform the isotopic analysis for iodine "until restored to within limits" is not included. This changes the CTS by deleting the explicit statement to perform the isotopic analysis for iodine until the limits are met.

The purpose of the statement "until the specific activity of the primary coolant is restored to within its limits," in CTS 3.4.8 (MODES 1, 2, 3, and 4) ACTION a., Table 4.4-4 Item 3.a, and footnote #, is to ensure the Surveillance is performed until the limit is met. In ITS, stating that the analysis is required until the specific activity is within limits is unnecessary. ITS LCO 3.0.2 requires the Required Actions of the entered Condition(s) to be performed upon discovery of failure to meet the LCO, until the LCO is met. If the LCO is met or is no longer applicable prior to the expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated. This change is acceptable since ITS LCO 3.0.2 will require the Required Action to be performed until the LCO is met. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS Table 4.4-4 Item 3.b Frequency requires, that one sample between 2 and 6 hours following a THERMAL POWER change exceeding (i.e., >) 15% of the RATED THERMAL POWER within a one hour period. ITS SR 3.4.16.2 Frequency requires, one sample between 2 and 6 hours after a THERMAL POWER change of greater than or equal to (i.e., ≥) 15% RTP within 1 hour. This changes the CTS by requiring one sample to be taken between 2 and 6 hours after a THERMAL POWER change of greater than or equal to (i.e., ≥) 15% RTP within 1 hour.

The purpose of CTS Table 4.4-4 Item 3.b Frequency is to ensure one sample is taken between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period. This change is acceptable because ITS SR 3.4.16.2 requires essentially the same Frequency with a sample between 2 and 6 hours after a THERMAL POWER change of greater than or equal to 15% RTP within 1 hour. This change is designated as administrative because it does not result in technical changes to CTS.

DISCUSSION OF CHANGES
ITS 3.4.16, RCS SPECIFIC ACTIVITY

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 4.4-4, Item 2 requires an isotopic analysis to determine whether DOSE EQUIVALENT I-131 concentration is within limit. CTS Table 4.4-4, Item 3 requires an isotopic analysis for iodine including I-131, I-132, I-133, I-134 and I-135. ITS SR 3.4.16.2 requires the verification that reactor coolant DOSE EQUIVALENT I-131 specific activity is within limit. ITS 3.4.16 Required Action A.1 requires the verification that DOSE EQUIVALENT I-131 is within the acceptable region. This changes the CTS by moving the detail that an isotopic analysis must be performed to satisfy the requirements of the Surveillances to the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS SR 3.4.16.2 and ITS 3.4.16 Required Action A.1 still retain the requirements to verify reactor coolant DOSE EQUIVALENT I-131 is within limit. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

3.4.8 LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.8.a 3.4.8.b A. DOSE EQUIVALENT I-131 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
3.4.8.a	A.1 Verify DOSE EQUIVALENT I-131 ≤ ~~[~~60~~]~~ µCi/gm.	Once per 4 hours
	AND	
3.4.8.b	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
3.4.8.d 3.4.8.d Note B. DOSE EQUIVALENT XE-133 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
3.4.8.d	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours

2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > [60] µCi/gm.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

3.4.8.c
3.4.8.d
3.4.8.c
2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	~~NOTE~~ ~~Only required to be performed in MODE 1.~~	
	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ [280] µCi/gm. (518.9)	[~~7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]

4.4.8
Table 4.4-4
Item 1.
3
3.4.8.b
2
Table 4.4-4
Item 1.
2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.8 Table 4.4-4 Item 2.	SR 3.4.16.2	----------NOTE---------- Only required to be performed in MODE 1. ----------		
3.4.8.a		Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ ~~[~~1.0~~]~~ µCi/gm.	~~[14 days~~ ~~OR~~	2
Table 4.4-4 Item 1. and 2.			In accordance with the Surveillance Frequency Control Program ~~]~~ AND	2
Table 4.4-4 Item 3.b			Between 2 and 6 hours after THERMAL POWER change of ≥ 15% RTP within a 1 hour period	

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

3.4.8 LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.4.8.a 3.4.8.b A. DOSE EQUIVALENT I-131 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
3.4.8.a	A.1 Verify DOSE EQUIVALENT I-131 ≤ [~~60~~] µCi/gm.	Once per 4 hours
	AND	
3.4.8.b	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
3.4.8.d 3.4.8.d Note B. DOSE EQUIVALENT XE-133 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
3.4.8.d	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours

2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > [60] µCi/gm.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

3.4.8.c
3.4.8.d
3.4.8.c
2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	~~NOTE~~ ~~Only required to be performed in MODE 1.~~	
	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ [280] µCi/gm. (518.9)	[~~7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program]

4.4.8
Table 4.4-4
Item 1.
3
3.4.8.b
2
Table 4.4-4
Item 1.
2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.4.8 Table 4.4-4 Item 2.	SR 3.4.16.2	----------NOTE---------- Only required to be performed in MODE 1. ----------		
3.4.8.a		Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ ~~[~~1.0~~]~~ μCi/gm.	~~[14 days~~ ~~OR~~	2
Table 4.4-4 Item 1. and 2.			In accordance with the Surveillance Frequency Control Program ~~]~~	2
			AND	
Table 4.4-4 Item 3.b			Between 2 and 6 hours after THERMAL POWER change of ≥ 15% RTP within a 1 hour period	

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.16, RCS SPECIFIC ACTIVITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ISTS SR 3.4.16 Note is deleted. PSL retains the current licensing basis that samples DOSE EQUIVALENT XE-133 in the applicable Modes.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in [10 CFR 100.11][10 CFR 50.67] (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents. 2

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (SR) (Ref. 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of [1 gpm] exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of [0.1] μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.19, "Secondary Specific Activity." 2

0.25 gpm per SG; 0.5 gpm total

2

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at [1.0] μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of 335), respectively. The second case assumes the initial reactor coolant iodine activity at [60.0] μCi/gm DOSE EQUIVALENT 2

2

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel damage is assumed for the SLB event and the activity released exceeds that released by the two iodine spike cases; therefore, the two iodine spike cases are not analyzed for the SLB event.

I-131 due to an iodine spike caused by a reactor or an RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be ~~[280]~~ µCi/gm DOSE EQUIVALENT XE-133. 518.9 1 2

The SGTR analysis assumes a rise in pressure in the ruptured SG causes radioactively contaminated steam to discharge to the atmosphere through the atmospheric dump valves or the main steam safety valves. The atmospheric discharge stops when the turbine bypass to the condenser removes the excess energy to rapidly reduce the RCS pressure and close the valves. The unaffected SG removes core decay heat by venting steam until the cooldown ends and the Shutdown Cooling (SDC) System is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SG removes core decay heat by venting steam to the atmosphere until the cooldown ends and the SDC System is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed ~~[~~60.0~~]~~ µCi/gm for more than 48 hours. 2

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The iodine specific activity in the reactor coolant is limited to ~~[~~1.0~~]~~ µCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to ~~[280]~~ µCi/gm DOSE EQUIVALENT XE-133. 518.9 2 2

The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is ≤ ~~[~~60.0~~]~~ µCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend. 2

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

BASES

ACTIONS (continued)

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODES(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

C.1 and C.2

If the Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > ~~[~~60.0~~]~~ µCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

2

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant ~~at least once every 7 days~~. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in the noble gas specific activity.

1

1

The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control. Program, and

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. ~~The 7 day Frequency considers the low probability of a gross fuel failure during this time.~~

1

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

~~A Note modifies the SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.~~

3

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The ~~14 day Frequency~~ is adequate to trend changes in the iodine activity level~~, considering noble gas activity is monitored every 7 days~~. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program, and

1

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

REFERENCES

~~Reviewer's Note~~

~~The first listed References 1 and 2 are for plants that are licensed to 10 CFR 100.11. The second set of References are for plants that are licensed to 10 CFR 50.67.~~

4

~~[1. 10 CFR 100.11.~~

2

~~2. Standard Review Plan (SRP) Section 15.1.5 Appendix A (SLB) and Section 15.6.3 (SGTR).~~

1. 10 CFR 50.67.

2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms." ~~]~~

2

3. FSAR, Section ~~[15.1.5]~~. 15.4.6

U

1 2

4. FSAR, Section ~~[15.6.3]~~. 15.4.4

U

1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in [10 CFR 100.11][10 CFR 50.67] (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents. 2

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (SR) (Ref. 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of [1 gpm] exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of [0.1] µCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.19, "Secondary Specific Activity." 2

0.25 gpm per SG; 0.5 gpm total

2

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at [1.0] µCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of 335), respectively. The second case assumes the initial reactor coolant iodine activity at [60.0] µCi/gm DOSE EQUIVALENT 2

2

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel damage is assumed for the SLB event and the activity released exceeds that released by the two iodine spike cases; therefore, the two iodine spike cases are not analyzed for the SLB event.

I-131 due to an iodine spike caused by a reactor or an RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be ~~[280]~~ µCi/gm DOSE EQUIVALENT XE-133. 518.9 1 2

The SGTR analysis assumes a rise in pressure in the ruptured SG causes radioactively contaminated steam to discharge to the atmosphere through the atmospheric dump valves or the main steam safety valves. The atmospheric discharge stops when the turbine bypass to the condenser removes the excess energy to rapidly reduce the RCS pressure and close the valves. The unaffected SG removes core decay heat by venting steam until the cooldown ends and the Shutdown Cooling (SDC) System is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SG removes core decay heat by venting steam to the atmosphere until the cooldown ends and the SDC System is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed ~~[~~60.0~~]~~ µCi/gm for more than 48 hours. 2

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The iodine specific activity in the reactor coolant is limited to ~~[~~1.0~~]~~ µCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to ~~[280]~~ µCi/gm DOSE EQUIVALENT XE-133. 518.9 2 2
The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is ≤ ~~[~~60.0~~]~~ µCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend. 2

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

BASES

ACTIONS (continued)

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODES(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

C.1 and C.2

If the Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > ~~[~~60.0~~]~~ µCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

2

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant ~~at least once every 7 days~~. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in the noble gas specific activity.

1

1

The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. ~~The 7 day Frequency considers the low probability of a gross fuel failure during this time.~~

1

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

~~A Note modifies the SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.~~

3

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The ~~14 day~~ Frequency is adequate to trend changes in the iodine activity level~~, considering noble gas activity is monitored every 7 days~~. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program, and

1

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

REFERENCES

~~Reviewer's Note~~

~~The first listed References 1 and 2 are for plants that are licensed to 10 CFR 100.11. The second set of References are for plants that are licensed to 10 CFR 50.67.~~

4

~~[1. 10 CFR 100.11.~~

2

~~2. Standard Review Plan (SRP) Section 15.1.5 Appendix A (SLB) and Section 15.6.3 (SGTR).~~

1. 10 CFR 50.67.

2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms." ~~]~~

2

3. FSAR, Section ~~[15.1.5]~~. 15.6.2

U

1 2

4. FSAR, Section ~~[~~15.6.3~~]~~.

U

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.16, BASES, RCS SPECIFIC ACTIVITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ISTS SR 3.4.16 Note is deleted. PSL retains the current licensing basis that samples DOSE EQUIVALENT XE-133 in the applicable Modes.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.16, RCS SPECIFIC ACTIVITY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 17

3.4.17, Steam Generator (SG) Tube Integrity

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTOR COOLANT SYSTEM

(RCS)

STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.17

~~3.4.5~~ SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube ~~repair~~ plugging criteria shall be plugged in accordance with the SG Program.

Applicability

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: *

ACTION A

~~a.~~ With one or more SG tubes satisfying the tube ~~repair~~ plugging criteria and not plugged in accordance with the Steam Generator Program;

~~1.~~ Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and

~~2.~~ Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering ~~HOT SHUTDOWN~~ MODE 4 following the next refueling outage or SG tube inspection.

ACTION B

~~b.~~ With the requirements and associated allowable outage time of Action a above not met or SG tube integrity not maintained, be in ~~HOT STANDBY~~ MODE 3 within 6 hours and in ~~COLD SHUTDOWN~~ MODE 5 within the next 30 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

~~4.4.5.1~~ Verify SG tube integrity in accordance with the Steam Generator Program.

SR 3.4.17.2

~~4.4.5.2~~ Verify that each inspected SG tube that satisfies the tube ~~repair~~ plugging criteria is plugged in accordance with the Steam Generator Program prior to entering ~~HOT SHUTDOWN~~ MODE 4 following a SG tube inspection.

ACTIONS Note

* Separate Action entry is allowed for each SG tube.

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~~**REACTOR COOLANT SYSTEM**~~

~~**SURVEILLANCE REQUIREMENTS** (Continued)~~

A01

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A01

REACTOR COOLANT SYSTEM

(RCS)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.17

~~3.4.5~~ SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube ~~repair~~ plugging criteria shall be plugged ~~or repaired~~ in accordance with the SG Program. ~~Repair applies only to the original SGs.~~

A02

Applicability

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:*

ACTION A

~~a.~~ With one or more SG tubes satisfying the tube ~~repair~~ plugging criteria and not plugged ~~(or repaired if original SGs)~~ in accordance with the Steam Generator Program;

A02

~~1.~~ Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and

~~2.~~ Plug ~~or repair~~ the affected tube(s) in accordance with the Steam Generator Program prior to entering ~~HOT SHUTDOWN~~ MODE 4 following the next refueling outage or SG tube inspection. ~~Repair applies only to the original SGs.~~

A02

A02

ACTION B

~~b.~~ With the requirements and associated allowable outage time of Action a above not met or SG tube integrity not maintained, be in ~~HOT STANDBY~~ MODE 3 within 6 hours and in ~~COLD SHUTDOWN~~ MODE 5 within the following 30 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

~~4.4.5.1~~ Verify SG tube integrity in accordance with the Steam Generator Program.

SR 3.4.17.2

~~4.4.5.2~~ Verify that each inspected SG tube that satisfies the tube ~~repair~~ plugging criteria is plugged ~~or repaired~~ in accordance with the Steam Generator Program prior to entering ~~HOT SHUTDOWN~~ MODE 4 following a SG tube inspection. ~~Repair applies only to the original SGs.~~

A02

ACTIONS Note * Separate Action entry is allowed for each SG tube

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DISCUSSION OF CHANGES
ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 2 only:** CTS 3.4.5 LCO, Action a., and 4.4.5.2 state, in part, that SG repair applies only to the original SGs. Unit 2 contains two AREVA Model 86/19Ti replacement Steam Generators (containing Alloy 690TT Tubing) which were installed at RFO-17 in 2007 (EC 235453, Rev. 0. “PCM-05137 Replacement Steam Generators 2A & 2B). Therefore, the details associated with repairing of SG tubes rather than plugging of SG tubes are deleted from CTS 3.4.5.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~18~~ 17 Steam Generator (SG) Tube Integrity

3

LCO 3.4.5

LCO 3.4.~~18~~ 17 SG tube integrity shall be maintained.

3

AND

All SG tubes satisfying the tube plugging ~~[or repair]~~ criteria shall be plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.

2

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

----------NOTE----------

Action Footnote *

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.1 A. One or more SG tubes satisfying the tube plugging ~~[or repair]~~ criteria and not plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection. AND	7 days 2
Action a.2	A.2 Plug ~~[or repair]~~ the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection 2
Action b. B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
SR 4.4.5.1	SR 3.4.~~18~~ 17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program	3
SR 4.4.5.2	SR 3.4.~~18~~ 17.2	Verify that each inspected SG tube that satisfies the tube plugging ~~[or repair]~~ criteria is plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection	3 2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~18~~ 17 Steam Generator (SG) Tube Integrity

3

LCO 3.4.5

LCO 3.4.~~18~~ 17 SG tube integrity shall be maintained.

3

AND

All SG tubes satisfying the tube plugging ~~[or repair]~~ criteria shall be plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.

2

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

----------NOTE----------

Action Footnote *

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.1 A. One or more SG tubes satisfying the tube plugging ~~[or repair]~~ criteria and not plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days 2
	AND	
Action a.2	A.2 Plug ~~[or repair]~~ the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection 2
Action b. B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
OR	AND	
SG tube integrity not maintained.	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
SR 4.4.5.1	SR 3.4.~~18~~ 17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program	3
SR 4.4.5.2	SR 3.4.~~18~~ 17.2	Verify that each inspected SG tube that satisfies the tube plugging ~~[or repair]~~ criteria is plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection	3 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ISTS 3.4.17, “Special Test Exception (STE)-RCS Loops,” has not been adopted. Therefore, ISTS 3.4.18, “Steam Generator (SG) Tube Integrity,” is renumbered as ITS 3.4.17, “Steam Generator (SG) Tube Integrity.”

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~18~~ 17 Steam Generator (SG) Tube Integrity 3

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops – MODES 1 and 2," LCO 3.4.5, "RCS Loops – MODE 3," LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

5.5.6

Specification ~~5.5.8~~, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification ~~5.5.8~~, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification ~~5.5.8~~. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions. 1

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of ~~[1~~ gallon per minute~~] or is assumed to increase to [1 gallon per minute] as a result of accident induced conditions~~. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. ~~3~~) or the NRC approved licensing basis (e.g., a small fraction of these limits).

0.5 2

10 CFR 50.67 (Ref. 3), 4 1

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging ~~[or repair]~~ criteria be plugged ~~[or repaired]~~ in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging ~~[or repair]~~ criteria is ~~[repaired or]~~ removed from service by plugging. If a tube was determined to satisfy the plugging ~~[or repair]~~ criteria but was not plugged ~~[or repaired]~~, the tube may still have tube integrity. 2

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall ~~[and any repairs made to it]~~, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube. 2

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification ~~5.5.8~~, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria. 1

5.5.6

BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. ~~4~~ 5) and Draft Regulatory Guide 1.121 (Ref. ~~5~~ 6). 1

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed ~~[1 gpm per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage.]~~ 0.5 gpm total through all SGs and 0.25 gpm through any one SG. 2 The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging ~~[or repair]~~ criteria but were not plugged ~~[or repaired]~~ in accordance with the Steam Generator Program as required by SR 3.4.~~18~~.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging ~~[or repair]~~ criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged ~~[or repaired]~~ has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the

17 2 3 2 2

BASES

ACTIONS (continued)

estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged ~~[or repaired]~~ prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment. 2

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.~~18~~ 17.1 3

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging ~~[or repair]~~ criteria. Inspection scope (i.e., 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.~~18~~.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. ~~6~~). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification ~~5.5.8~~ contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification ~~5.5.8~~ until subsequent inspections support extending the inspection interval.

SR 3.4.~~18~~.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging ~~[or repair]~~ criteria is ~~[repaired or]~~ removed from service by plugging. The tube plugging ~~[or repair]~~ criteria delineated in Specification ~~5.5.8~~ are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging ~~[or repair]~~ criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

~~[Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.]~~

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging ~~[or repair]~~ criteria are plugged ~~[or repaired]~~ prior to subjecting the SG tubes to significant primary to secondary pressure differential.

BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.67.

4 ~~3~~. 10 CFR 100.

5 ~~4~~. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

6 ~~5~~. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

7 ~~6~~. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~18~~ 17 Steam Generator (SG) Tube Integrity 3

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops – MODES 1 and 2," LCO 3.4.5, "RCS Loops – MODE 3," LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

5.5.6

Specification ~~5.5.8~~, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification ~~5.5.8~~, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification ~~5.5.8~~. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions. 1

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analysis, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of ~~[1~~ gallon per minute~~] or is assumed to increase to [1 gallon per minute]~~ as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4. 16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. ~~3~~) or the NRC approved licensing basis (e.g., a small fraction of these limits).

0.5 2 1 10 CFR 50.67 (Ref. 3), 4

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging ~~[or repair]~~ criteria be plugged ~~[or repaired]~~ in accordance with the Steam Generator Program. 2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging ~~[or repair]~~ criteria is ~~[repaired or]~~ removed from service by plugging. If a tube was determined to satisfy the plugging ~~[or repair]~~ criteria but was not plugged ~~[or repaired]~~, the tube may still have tube integrity. 2

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall ~~[~~and any repairs made to it~~]~~, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube. 2

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification ~~5.5.8~~, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria. 1

5.5.6

BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. ~~4~~ 5) and Draft Regulatory Guide 1.121 (Ref. ~~5~~ 6). 1

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed ~~[1 gpm per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage.]~~ The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident. 2

0.5 gpm total through all SGs and 0.25 gpm through any one SG.

BASES

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging ~~[or repair]~~ criteria but were not plugged ~~[or repaired]~~ in accordance with the Steam Generator Program as required by SR 3.4.~~18~~.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging ~~[or repair]~~ criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged ~~[or repaired]~~ has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the

17
2
3
2
2

BASES

ACTIONS (continued)

estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged ~~[or repaired]~~ prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment. 2

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.~~18~~ 17.1 3

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging ~~[or repair]~~ criteria. Inspection scope (i.e., 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.~~18~~.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. ~~6~~). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification ~~5.5.8~~ contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification ~~5.5.8~~ until subsequent inspections support extending the inspection in~~v~~erval.

SR 3.4.~~18~~.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging ~~[or repair]~~ criteria is ~~[repaired or]~~ removed from service by plugging. The tube plugging ~~[or repair]~~ criteria delineated in Specification ~~5.5.8~~ are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging ~~[or repair]~~ criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

~~[Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.]~~

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging ~~[or repair]~~ criteria are plugged ~~[or repaired]~~ prior to subjecting the SG tubes to significant primary to secondary pressure differential.

BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.67.

4 → ~~3~~. 10 CFR 100.

5 → ~~4~~. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

6 → ~~5~~. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

7 → ~~6~~. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.17, BASES, STEAM GENERATOR (SG) TUBE INTEGRITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ISTS 3.4.17, “Special Test Exception (STE)-RCS Loops,” has not been adopted. Therefore, ISTS 3.4.18, “Steam Generator (SG) Tube Integrity,” is renumbered as ITS 3.4.17, “Steam Generator (SG) Tube Integrity.”

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.4.17, STEAM GENERATOR (SG) TUBE INTEGRITY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 18

Relocated/Deleted Current Technical Specifications

Unit 1 CTS 3/4.4.9.2, Pressurizer

Unit 2 CTS 3/4.4.9.2, Pressurizer Heatup/Cooldown Limits

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

R01

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum Reactor Coolant System spray water temperature differential of 350°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psia within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit in accordance with the Surveillance Frequency Control Program during steady state operation.

REACTOR COOLANT SYSTEM

PRESSURIZER HEATUP/COOLDOWN LIMITS

LIMITING CONDITION FOR OPERATION R01

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period, and

b. A maximum cooldown of 200°F in any 1-hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program during system heatup or cooldown.

DISCUSSION OF CHANGES
Unit 1 CTS 3/4.4.9.2, PRESSURIZER
UNIT 2 CTS 3/4.4.9.2, PRESSURIZER HEATUP/COOLDOWN LIMITS

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R01 The limitation on the pressurizer pressure and temperature (P/T) ensures that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements. As such, the Technical Specification places limits on variables consistent with structural analysis results. These limits do not represent an initial condition assumption of a DBA or transient. Although the limits represent operating restrictions, and Criterion 2 includes operating restrictions, the Criterion 2 discussion of the Final Policy Statement specified that only those operating restrictions required to preclude unanalyzed accidents and transients be included in Technical Specifications.

The ITS does not include these Specifications. This changes the CTS by relocating these Specification to the Technical Requirements Manual (TRM). This change is acceptable because the Pressurizer Temperature Specification does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. Pressurizer P/T limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary.

2. Pressurizer P/T limits do not represent a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Pressurizer P/T limits are not part of a primary success path in the mitigation of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Pressurizer P/T limits are not addressed in the PSL PSA and do not represent a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Since 10 CFR 50.36(c)(2)(ii) criteria have not been satisfied, the Unit 1 Pressurizer Specification, which specifies pressurizer heatup and cooldown limits, and Unit 2 Pressurizer Heatup and Cooldown Limits Specification may be

relocated to a licensee controlled document outside the Technical Specifications. Pressure and temperature limits associated with the limiting RCS pressure boundary components are retained in separate Technical Specifications. ASME code requirements associated with the limits on pressurizer heatup and cooldown will continue to be controlled pursuant 10 CFR 50.55a. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification does not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
Unit 1 CTS 3/4.4.9.2, PRESSURIZER
UNIT 2 CTS 3/4.4.9.2, PRESSURIZER HEATUP/COOLDOWN LIMITS

There are no specific No Significant Hazards Considerations for this Specification.

Unit 1 CTS 3/4.4.15, Reactor Coolant System Vents

Unit 2 CTS 3/4.4.10, Reactor Coolant System Vents

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

REACTOR COOLANT SYSTEM

3/4.4.15 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.15 At least one Reactor Coolant System vent path consisting of two vent valves and one block valve powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Pressurizer steam space, and

b. Reactor vessel head.

R01

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Reactor Coolant System vent paths inoperable, maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.15 Each Reactor Coolant System vent path shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each vent valve through at least one complete cycle of full travel from the control room.

3. Verifying flow through the Reactor Coolant System vent paths during venting.

REACTOR COOLANT SYSTEM

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.10 At least one Reactor Coolant System vent path consisting of two vent valves and one block valve powered from emergency buses shall be OPERABLE and closed at each of the following locations:

R01

a. Pressurizer steam space, and

b. Reactor vessel head.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Reactor Coolant System vent paths inoperable, maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.1 Each Reactor Coolant System vent path shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each vent valve through at least one complete cycle of full travel from the control room.

3. Verifying flow through the Reactor Coolant System vent paths during venting.

DISCUSSION OF CHANGES
Unit 1 CTS 3/4.4.15 – Unit 2 CTS 3/4.4.10, REACTOR COOLANT SYSTEM VENTS

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R01 Reactor Coolant System (RCS) vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The function, capabilities, and testing requirements of the RCS vent system are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

The purpose of the pressurizer safety valves is to provide primary relief capability to exhaust noncondensible gases and steam from the RCS during a DBA or transient at power and the pressurizer power operated relief valves (PORVs) provide vent capability during RCS low temperature and pressure conditions.

The ITS does not include these Specifications. This changes the CTS by relocating these Specifications to the Technical Requirements Manual (TRM). This change is acceptable because the Reactor Coolant System Vent Specification does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The RCS vents do not function as installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The RCS vents are not process variables, design features, or operating restrictions that represents an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The RCS vents are not a structures, systems or components that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The RCS vents are not specifically addressed in the PSL PSA and do not represent components which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Since 10 CFR 50.36(c)(2)(ii) criteri have not been satisfied, the Reactor Coolant System Vents Specification may be relocated to a licensee controlled document

outside Technical Specifications. Pressure relief requirements are retained in separate Technical Specifications and ensure relief capability is provided to exhaust noncondensible gases and steam from the primary system. In addition, this NUREG-0737 requirement is required pursuant to 10 CFR 50.34(f)(2)(vi). Compliance with applicable portions of 10 CFR 50.34(f)(2) is required by the operating licenses of PSL Units 1 and 2. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification does not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
Unit 1 CTS 3/4.4.15 – Unit 2 CTS 3/4.4.10, REACTOR COOLANT SYSTEM VENTS

There are no specific No Significant Hazards Considerations for this Specification.

Unit 2 CTS 3/4.10.3, Reactor Coolant Loops - Special Test Exceptions

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

M01

SPECIAL TEST EXCEPTIONS

3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1 and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater then 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during startup and PHYSICS TESTS.

4.10.3.2 Each wide range logarithmic and power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

DISCUSSION OF CHANGES
CTS 3/4.10.3, REACTOR COOLANT LOOPS – UNIT 2 ONLY

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M01 Current Technical Specifications (CTS) 3/4.10.3 provides an exception to the limitations of CTS 3.4.1 and the noted requirements of CTS Tables 2.2-1 and 3.3-1. This special test exception permits reactor criticality under reduced flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels. The Improved Technical Specifications (ITS) do not contain this special test exception. This changes the CTS by eliminating a special test exception.

This change is acceptable because this type of PHYSICS TEST is no longer performed. Reactor Coolant System (RCS) flow will be maintained in accordance with the requirements of LCO 3.4.4, "RCS Loops - MODES 1 and 2," during PHYSICS TESTS in MODES 1 and 2. As a result, this CTS Special test exception is not needed. This change is designated as more restrictive because an exception to the CTS is being deleted.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.10.3, REACTOR COOLANT LOOPS – UNIT 2 ONLY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 19

ISTS Not Adopted

ISTS 3.4.17, Special Test Exception – RCS Loops

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Special Test Exception (STE)-RCS Loops

1

LCO 3.4.17 The requirements of LCO 3.4.4, "RCS Loops - MODES 1 and 2," and the listed requirements of LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating," for the [(Analog) RC flow low, thermal margin or low pressure, and asymmetric steam generator transient protective trip functions] [(Digital) high log power, high local power density, low departure from nucleate boiling ratio protective trip functions] may be suspended provided:

a. THERMAL POWER ≤ 5% RTP and

b. The reactor trip setpoints of the OPERABLE power level channels are set ≤ 20% RTP.

APPLICABILITY: MODE 2, during startup and PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open reactor trip breakers.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify THERMAL POWER ≤ 5% RTP.	[1 hour OR In accordance with the Surveillance Frequency Control Program]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.17.2	Perform a CHANNEL FUNCTIONAL TEST on each logarithmic power level and linear power level neutron flux monitoring channel.	12 hours prior to initiating startup or PHYSICS TESTS

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.17, SPECIAL TEST EXCEPTION (STE) - RCS LOOPS

1. This Reactor Coolant System Loops - Test Exceptions Specification is not included in the PSL Unit 1 and Unit 2 ITS because the exception is not needed to perform any required startup or PHYSICS TESTS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Special Test Exception (STE) RCS Loops

1

BASES

BACKGROUND

This special test exception to LCO 3.4.4, "RCS Loops - MODES 1 and 2," and LCO 3.3.1, "RPS Instrumentation," permits reactor criticality under no flow conditions during PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) while at low THERMAL POWER levels. Section XI of 10 CFR Part 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant as specified in 10 CFR 50, Appendix A, GDC 1 (Ref. 2).

The key objectives of a test program are to provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the facility has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations.

The tests will include verifying the ability to establish and maintain natural circulation following a plant trip between 10% and 20% RTP, performing natural circulation cooldown on emergency power, and during the cooldown, showing that adequate boron mixing occurs and that pressure can be controlled using auxiliary spray and pressurizer heaters powered from the emergency power sources.

APPLICABLE SAFETY ANALYSES

As described in LCO 3.0.7, compliance with Special Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

BASES

LCO

This LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without this LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODES 1 and 2), and the appropriate tests could not be performed. 1

In MODE 2, where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER is < 5% RTP and the reactor trip setpoints of the OPERABLE power level channels are set ≤ 20% RTP. These limits ensure no Safety Limits or fuel design limits will be violated.

The exception is allowed even though there are no bounding safety analyses. These tests are allowed since they are performed under close supervision during the test program and provide valuable information on the plant's capability to cool down without offsite power available to the reactor coolant pumps.

APPLICABILITY

This LCO ensures that the plant will not be operated in MODE 1 without forced circulation. It only allows testing under these conditions while in MODE 2. This testing establishes that heat input from nuclear heat does not exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.

ACTIONS

A.1

If THERMAL POWER increases to > 5% RTP, the reactor must be tripped immediately. This ensures the plant is not placed in an unanalyzed condition and prevents exceeding the specified acceptable fuel design limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

THERMAL POWER must be verified to be within limits to ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. [The hourly Frequency has been shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation is within the LCO limits. Plant operations are conducted slowly during the performance of PHYSICS TESTS, and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.

BASES

SURVEILLANCE REQUIREMENTS (continued)

1

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------REVIEWER'S NOTE----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

SR 3.4.17.2

Within 12 hours of initiating startup or PHYSICS TESTS, a CHANNEL FUNCTIONAL TEST must be performed on each logarithmic power level and linear power level neutron flux monitoring channel to verify OPERABILITY and adjust setpoints to proper values. This will ensure that the Reactor Protection System is properly aligned to provide the required degree of core protection during startup or the performance of the PHYSICS TESTS. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The interval is adequate to ensure that the appropriate equipment is OPERABLE prior to the tests to aid the monitoring and protection of the plant during these tests.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50, Appendix A, GDC 1, 1988.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.17, BASES, SPECIAL TEST EXCEPTION (STE) - RCS LOOPS

1. Changes have been made to be consistent with changes made to the ISTS.

ENCLOSURE 2

VOLUME 10

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.5
EMERGENCY CORE COOLING SYSTEMS (ECCS)

Revision 1

LIST OF ATTACHMENTS

1. 3.5.1, Safety Injection Tanks (SITs)
2. 3.5.2, ECCS - Operating
3. 3.5.3, ECCS - Shutdown
4. 3.5.4, Refueling Water Tank (RWT)
5. 3.5.5, Trisodium Phosphate (TSP) – Unit 2 only

ATTACHMENT 1

3.5.1, Safety Injection Tanks (SITs)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

SAFETY INJECTION TANKS (SITs)

LIMITING CONDITION FOR OPERATION

LCO 3.5.1 3.5.1 ~~Each reactor coolant system~~ Four safety injection tank s shall be OPERABLE ~~with:~~

SR 3.5.1.1 ~~a.~~ ~~The~~ Verify each SIT isolation valve is fully open~~,~~

SR 3.5.1.2 ~~b.~~ ~~Between~~ Verify borated water volume in each SIT is ≥ 1090 and ≤ 1170 cubic feet ~~of borated water,~~

SR 3.5.1.4 ~~c.~~ ~~A minimum~~ Verify boron concentration in each SIT is ≥ ~~of~~ 1900 ppm ≤ 2200 ppm~~,~~ and M01

SR 3.5.1.3 ~~d.~~ ~~A~~ nitrogen cover-pressure in each SIT is ≥ ~~of between~~ 230 and ≤ 280 psig.

Applicability **APPLICABILITY:** MODES 1, 2 and 3 with pressurizer pressure ≥ 1750 psia

ACTION:

Condition A ~~a.~~ ~~With~~ one SIT inoperable due to boron concentration level not within limits, or one SIT inoperable due to ~~an~~ inability to verify ~~the required water volume~~ or ~~nitrogen cover-~~pressure, restore ~~the inoperable~~ SIT to OPERABLE status with 72 hours; otherwise, be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ 6 hours.

Required Action A.1 → MODE 3

Required Action C .1, C.2 → 12

Reduce pressurizer pressure to < 1750 psia — A02

Condition B ~~b.~~ ~~With~~ one SIT inoperable ~~due to~~ for reasons other than ~~those stated in~~ ACTION-a, restore ~~the inoperable~~ SIT to OPERABLE status within 24 hours ; otherwise, be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ 6 hours.

Required Action B.1 → MODE 3

Required Action C .1, C.2 → 12

Reduce pressurizer pressure to < 1750 psia — A02

Add proposed Action D — A03

SURVEILLANCE REQUIREMENTS

~~4.5.1~~ ~~Each safety injection tank shall be demonstrated OPERABLE:~~

SR 3.5.1.1, SR 3.5.1.2, SR 3.5.1.3 ~~a.~~ In accordance with the Surveillance Frequency Control Program ~~by:~~

SR 3.5.1.2 SR 3.5.1.3 ~~1.~~ Verify~~ing that the~~ borated water volume and nitrogen cover-pressure in each SIT is ~~in the tanks are~~ within ~~their~~ limits~~, and~~

SR 3.5.1.1 ~~2.~~ Verify~~ing that~~ each safety injection tank isolation valve is open.

A01

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

NOTE Only required to be performed for affected SIT.

A04

SR 3.5.1.4

~~b.~~ In accordance with the Surveillance Frequency Control Program and once within 6 hours after each solution volume increase of ≥ 1% of tank volume ~~by~~ verify~~ing the~~ boron concentration ~~of the safety injection tank solution~~. ~~This latter surveillance is not required when the volume increase makeup source is the RWT and the RWT has not been diluted since verifying that the RWT boron concentration is equal to or greater than the safety injection tank boron concentration limit~~.

that is not the result of addition from the RWT.

in each SIT is ≥ 1900 ppm and ≤ 2200 ppm.

LA01

SR 3.5.1.5

~~c.~~ In accordance with the Surveillance Frequency Control Program ~~when the RCS pressure is above 1750 psia, by~~ verify~~ing that~~ power ~~to the~~ isolation valve operator ~~is removed~~ ~~by maintaining the breaker open under administrative control~~.

removed from each SIT

A05

LA02

~~d. In accordance with the Surveillance Frequency Control Program by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:~~

~~1. When the RCS pressure exceeds 350 psia, and~~

~~2. Upon receipt of a safety injection test signal.~~

LA04

A01

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS (SITs)

LIMITING CONDITION FOR OPERATION

LCO 3.5.1 3.5.1 ~~Each Reactor Coolant System~~ Four safety injection tank s shall be OPERABLE ~~with:~~

SR 3.5.1.1 ~~a.~~ ~~The~~ Verify each SIT isolation valve is fully open.

SR 3.5.1.2 ~~b.~~ ~~A contained~~ Verify borated water volume in each SIT is ≥ 1420 and ≤ 1556 cubic feet~~,~~

SR 3.5.1.4 ~~c.~~ ~~A~~ Verify boron concentration in each SIT is ≥ ~~of between~~ 1900 and ppm ≤ 2200 ppm ~~of boron, and~~

SR 3.5.1.3 ~~d.~~ ~~A~~ nitrogen cover-pressure in each SIT is ≥ ~~of between~~ 500 and ≤ 650 psig.

> **~~NOTE~~**
>
> ~~When in MODE 3 with pressurizer pressure is less than 1750 psia, at least three safety injection tanks shall be OPERABLE, each with a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 1250 and 1556 cubic feet with a boron concentration of between 1900 and 2200 ppm of boron. With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 833 and 1556 cubic feet with a boron concentration of between 1900 and 2200 ppm of boron.~~

LA03

Applicability **APPLICABILITY:** MODES 1, 2 and 3 with pressurizer pressure ≥ 1750 psia.

ACTION:

Condition A ~~a.~~ ~~With~~ one SIT inoperable ~~due~~ to level boron concentration not within limits, or one SIT inoperable due to ~~an~~ inability to verify ~~the required~~ water volume or ~~nitrogen cover-~~pressure, restore ~~the inoperable~~ SIT to OPERABLE status with 72 hours; otherwise, be in ~~at least HOT~~ STANDBY within ~~the next~~ 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ 6 hours.

Required Action A.1 MODE 3

Required Action C .1, C.2 12

Reduce pressurizer pressure to < 1750 psia — A02

Condition B ~~b.~~ ~~With~~ one SIT inoperable ~~due to~~ for reasons other than ~~those stated in~~ ACTION-a, restore ~~the inoperable~~ SIT to OPERABLE status within 24 hours; otherwise, be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ 6 hours.

Required Action B.1 MODE 3

Required Action C .1, C.2 12

Reduce pressurizer pressure to < 1750 psia — A02

Add proposed Action D — A03

SURVEILLANCE REQUIREMENTS

~~4.5.1.1~~ ~~Each safety injection tank shall be demonstrated OPERABLE:~~

SR 3.5.1.1, SR 3.5.1.2, SR 3.5.1.3 ~~a.~~ In accordance with the Surveillance Frequency Control Program ~~by:~~

SR 3.5.1.2 SR 3.5.1.3 ~~1.~~ Verify~~ing that the~~ borated water volume and nitrogen cover-pressure in each SIT is ~~in the tanks are~~ within ~~their~~ limits~~, and~~

SR 3.5.1.1 ~~2.~~ Verify~~ing that~~ each safety injection tank isolation valve is open.

A01

MERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

NOTE Only required to be performed for affected SIT. — A03

SR 3.5.1.4

~~b.~~ In accordance with the Surveillance Frequency Control Program and once within 6 hours after each solution volume increase of ~~greater than or equal to~~ 1% of tank volume ~~by~~ verify~~ing the~~ boron concentration ≥ in each SIT is ≥ 1900 ppm and ≤ 2200 ppm. ~~of the safety injection tank solution~~. that is not the result of addition from the RWT. ~~This latter surveillance is not required when the volume increase makeup source is the RWT and the RWT has not been diluted since verifying that the RWT boron concentration is equal to or greater than the safety injection tank boron concentration limit~~.

LA01

SR 3.5.1.5

~~c.~~ In accordance with the Surveillance Frequency Control Program ~~when the RCS pressure is above 700 psia, by~~ verify~~ing that~~ power ~~to the~~ isolation valve operator ~~is disconnected~~ removed from each SIT ~~by maintaining the breaker open by administrative controls~~.

LA05

LA02

~~d. In accordance with the Surveillance Frequency Control Program by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:~~

~~1. When an actual or simulated RCS pressure signal exceeds 515 psia, and~~

~~2. Upon receipt of a safety injection test signal.~~

LA04

DISCUSSION OF CHANGES
ITS 3.5.1, SAFETY INJECTION TANKS (SITs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.5.1 Actions a. and b. require in part, “otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours” when a SIT cannot be restored to OPERABLE status within the required time. ITS 3.5.1 Required Action C.2 requires, when the Required Actions and associated Completion Times are not met, reducing pressurizer pressure to < 1750 psia. This changes the end state in CTS 3.5.1 Actions a. and b. from MODE 4 to MODE 3 with pressurizer pressure < 1750 psia.

The purpose of CTS 3.5.1 Action a. and b. is to place the plant in a condition in which the equipment is no longer required. CTS 3.5.1 Action a. and b. provide appropriate actions to take for an inoperable SIT when the unit is in MODE 1, 2, or 3 with pressurizer pressure is ≥ 1750 psia. The SIT is not required to be OPERABLE when pressurizer pressure is < 1750 psia. Below 1750 psia, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below 10 CFR 50.46 limit of 2200°F. Therefore, in accordance with CTS 3.0.2 (ITS LCO 3.0.2), when pressurizer pressure is < 1750 psia, the requirements to be in MODE 4 in CTS 3.5.1 Action a. and b. are no longer required to be completed. The SITs, with pressurizer pressure below 1750 psia, are no longer required to be OPERABLE.

These changes are designated as administrative changes and are acceptable because they maintain the current requirement to place the unit in a condition in which the SITs are no longer required consistent with the requirements of CTS 3.0.2 (ITS LCO 3.0.2) and, therefore, do not result in a technical change to the CTS.

A03 CTS 3.5.1 does not contain a specific ACTION for two or more SITs inoperable. With two or more SITs inoperable, CTS 3.0.3 would be entered. ITS 3.5.1 ACTION D directs entry into LCO 3.0.3 when two or more SITs are inoperable. This changes the CTS by specifically stating to enter LCO 3.0.3 in this System Specification.

This change is acceptable because the action taken with two or more SITs inoperable is unchanged. Adding this ACTION is consistent with the ITS convention of directing entry into LCO 3.0.3 when multiple ACTIONS are presented in the ITS, and entry into these multiple ACTIONS could result in a loss of safety function. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 Unit 1 CTS 4.5.1.b, and Unit 2 CTS 4.5.1.1.b state, in part, that verification of the boron concentration of the SIT solution is required once within 6 hours after each solution volume increase of ≥ 1% of tank volume; however, this surveillance is not required when the volume increase makeup source is the RWT and the RWT has not been diluted since verifying that the RWT boron concentration is equal to or greater than the SIT boron concentration limit.

The purpose of the Unit 1 CTS 4.5.1.b, and Unit 2 CTS 4.5.1.1.b Surveillance Frequency is to designate the Frequency, as well as certain requirements or special conditions. ITS SR 3.5.1.4 adds a Note to the Surveillance Frequency that states, "Only required to be performed for affected SIT." The CTS Surveillance Frequency is implicit that the Surveillance be performed for only the affected SIT, that being the SIT subject to the solution volume increase. CTS does not contain this Note. This changes the CTS by adding the ITS SR 3.5.1.4 Note to clarify that the Surveillance is only required to be performed for the affected SIT. This change is acceptable because ITS SR 3.5.1.4 Note clarifies that the Surveillance is only required to be performed for the affected SIT. This change is designated as administrative because it does not result in technical changes to CTS.

A05 **Unit 1 only:** Unit 1 CTS 4.5.1.c requires verification that power be removed from each SIT isolation valve operator when RCS pressure is above 1750 psia. ITS SR 3.5.1.5 requires verification that power be removed from each SIT isolation valve operator when pressurizer pressure is greater than or equal to a specific pressure. This changes the CTS by deleting the specific pressure at which power be removed from the SIT isolation valve operators.

The purpose of the CTS requirement is to ensure power is removed from each SIT isolation valve operator when RCS pressure is above a specific value to further ensure that an active failure could not result in the undetected closure of an SIT motor operated isolation valve. This pressure value is typically above the pressure the SITs are required to be OPERABLE. For instance, the ISTS 3.5.1 Applicability is, in part, with pressurizer pressure ≥ 700 psia and ISTS SR 3.5.1.5 requires power to be removed from the isolation valve operator ≥ 2000 psia. Per the requirements of CTS 4.0.1 (ITS SR 3.0.1) and CTS 4.0.4 (ITS SR 3.0.4), SRs must be met prior to entry into the Applicability, which in the case of CTS 3.5.1 (ITS LCO 3.5.1) is MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia. Conversely, as stated in the ITS Bases of SR 3.0.1, Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable. Therefore, it is unnecessary to specifically require power to the valve operator be removed "when in MODE 3 with pressure ≥ 1750 psia" and this additional wording has been removed from Unit 1 ITS SR 3.5.1.5. As such, this additional wording has been removed from Unit 2 ITS SR 3.5.1.5.

This change is acceptable because ITS SR 3.5.1.5 requires essentially the same verification by requiring that power be removed from each SIT isolation valve operator when the SITs are required to be OPERABLE (i.e., when pressurizer pressure is greater than or equal to 1750 psia for Unit 1. This change is designated as administrative because it does not result in technical changes to CTS because the Surveillance requirement will continue to require power to the

SIT isolation valve operator to be removed when the SITs are required to be OPERABLE (i.e., in MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia).

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS 3.5.1.c requires a minimum SIT boron concentration of 1900 ppm but does not require a maximum boron concentration. ITS SR 3.5.1.4 requires verification that SIT boron concentration is within both a minimum and maximum limit. Similarly, Unit 2 CTS 3.5.1.c requires boron concentration within a minimum 1900 ppm and a maximum 2200 ppm. This changes the Unit 1 CTS by adding a SIT boron concentration maximum limit of 2200 ppm.

The purpose of CTS 3.5.1.c is to provide a SIT minimum boron concentration limit to ensure that, following a LOCA with a minimum level in the SIT, the reactor will remain subcritical in the cold condition following mixing of the SIT and RCS water volumes. The SIT maximum boron concentration limit is based on boron precipitation in the core following a LOCA which could result in reduced heat removal capability between the fuel cladding and the reactor coolant. This change is acceptable, because the maximum boron concentration limit provides assurance that the SIT is capable of performing its function when required. This change is designated as more restrictive because it adds a SIT maximum boron concentration limit to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* PSL Unit 1 CTS 4.5.1.b, and PSL Unit 2 CTS 4.5.1.1.b state that verification of boron concentration of the SIT solution is required once within 6 hours after each solution volume increase of ≥ 1% of tank volume, and that the Surveillance is not required when the volume increase makeup source is the RWT and the RWT has not been diluted since verifying that the RWT boron concentration is equal to or greater than the SIT boron concentration limit. ITS SR 3.5.1.4 states that verification of boron concentration of each SIT is required once within 6 hours after each solution volume increase of ≥ 1% of tank volume that is not the result of addition from the refueling water tank. ITS does not include a Surveillance Frequency requirement that retains the statement that the RWT has not been diluted since verifying that the RWT boron concentration is equal to or greater than the SIT boron concentration limit. This changes the CTS by relocation of the CTS requirement to the ITS Bases. The ITS 3.5.1.4 Bases will include that it is not necessary to verify boron concentration if the added water is from the RWT, because the boron concentration of the water contained in the RWT is within the SIT boron concentration requirements.

The removal of these details for performing surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS SR 3.5.1.4 retains the CTS PSL Unit 1 CTS 4.5.1.b, and PSL Unit 2 CTS 4.5.1.1.b, requirement by stating that verification of boron concentration of the SIT solution is required once within 6 hours after each solution volume increase of ≥ 1% of tank volume that is not the result of addition from the refueling water tank. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications. | R1

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* PSL Unit 1 CTS 4.5.1.c, and PSL Unit 2 CTS 4.5.1.1.c require verifying that power to the (SIT) isolation valve operator is removed / disconnected by maintaining the breaker open under administrative control. ITS SR 3.5.1.5 states verify power is removed from each SIT isolation valve operator. ITS does not include the detailed requirement that the breaker to the isolation valve operator be maintained open under administrative control. This changes the CTS by relocation of the administrative control details to the ITS Bases. The ITS 3.5.1.5 Bases will include that removal of power to the SIT isolation valve operators is conducted under administrative control.

The removal of these details for performing surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS SR 3.5.1.5 retains the requirement to ensure power is removed from the SIT isolation valve operator. ITS SR 3.5.1.5 states verify power is removed from each SIT isolation valve operator. This change is acceptable because these types of procedural details will continue to be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA03 **Unit 2 only:** *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 3.5.1 LCO Note states “When in MODE 3 with pressurizer pressure less than 1750 psia, at least three safety injection tanks shall be OPERABLE, each with a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 1250 and 1556 cubic feet with a boron concentration of between 1900 and 2200 ppm of boron. With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 833 and 1556 cubic feet with a boron

concentration of between 1900 and 2200 ppm of boron." ITS 3.5.1 does not include this requirement. PSL Unit 2 CTS 3.5.1 Applicability is MODES 1, 2, and 3 with pressurizer pressure ≥ 1750 psia. The SIT is not required to be OPERABLE when pressurizer pressure is < 1750 psia. Therefore, in accordance with CTS 3.0.2 (ITS LCO 3.0.2), when pressurizer pressure is < 1750 psia, the requirements in the Note are no longer required. Similarly, DOC A02 changes the end state in CTS 3.5.1 Actions a. and b. from MODE 4 to MODE 3 with pressurizer pressure < 1750 psia. This changes the CTS by relocating the requirements in the Note to the Technical Requirements Manual.

The removal of SIT requirements when in MODE 3 with pressurizer pressure < 1750 psia is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. The ITS retains the requirements for SITs to be OPERABLE when in MODES 1, 2, and 3 with pressurizer pressure ≥ 1750 psia. When pressurizer pressure is below 1750 psia the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 limit of 2200°F. Therefore, the SITs are not required to be OPERABLE, and the SIT motor operated isolation valves may be closed to isolate the SITs from the RCS during plant shutdown or startup. Also, this change is acceptable because these types of procedural details will continue to be controlled in the TRM. Changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because these SIT requirements are not required by the CTS 3.5.1 and ITS 3.5.1 Applicability, that being MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia. | R1

LA04 *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* PSL Unit 1 CTS 4.5.1.d states "In accordance with the Surveillance Frequency Control Program by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions: When the RCS pressure exceeds 350 psia, and Upon receipt of a safety injection test signal." PSL Unit 2 CTS 4.5.1.1.d states "In accordance with the Surveillance Frequency Control Program by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions: When an actual or simulated RCS pressure signal exceeds 515 psia, and Upon receipt of a safety injection test signal.

ITS 3.5.1 does not include this requirement. ITS SR 3.5.1.1 requires verification that each SIT isolation valve is fully open when in MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia. Additionally, ITS SR 3.5.1.5 requires verification that power is removed from each SIT isolation valve operator when pressurizer pressure is ≥ 1750 psia. ITS SR 3.5.1.1 and SR 3.5.1.5 provide assurance each SIT isolation valve is open and remains open (power removed from its valve operator) when in MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia. | R1

The SITs are not required to be OPERABLE when pressurizer pressure is < 1750 psia. Therefore, in accordance with CTS 3.0.2 (ITS LCO 3.0.2), when pressurizer pressure is < 1750 psia, the requirements in PSL Unit 1 CTS 4.5.1.d

and PSL Unit 2 CTS 4.5.1.1.d are no longer required. Similarly, DOC A02 changes the end state in CTS 3.5.1 Actions a. and b. from MODE 4 to MODE 3 with pressurizer pressure < 1750 psia. This changes the CTS by relocating the requirements in the PSL Unit 1 CTS 4.5.1.d and PSL Unit 2 CTS 4.5.1.1.d.

The removal of SIT requirements when in MODE 3 with pressurizer pressure < 1750 psia is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. The ITS retains the requirements for SITs to be OPERABLE when in MODES 1, 2, and 3 with pressurizer pressure ≥ 1750 psia. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because these SIT requirements are not required by the CTS 3.5.1 and ITS 3.5.1 Applicability, that being MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia. | R1

LA05 **Unit 2 only:** *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* Unit 2 CTS 4.5.1.1.c requires verification that the power be removed from each SIT isolation valve operator when RCS pressure is above 700 psia. ITS SR 3.5.1.5 requires verification that power be removed from each SIT isolation valve operator. This changes the CTS by deleting the specific pressure at which power be removed from the SIT isolation valve operators. | R1

The purpose of the CTS requirement is to ensure power is removed from each SIT isolation valve operator when RCS pressure is above a specific value to further ensure that an active failure could not result in the undetected closure of an SIT motor operated isolation valve. This pressure value is typically above the pressure the SITs are required to be OPERABLE. For instance, the ISTS 3.5.1 Applicability is, in part, with pressurizer pressure ≥ 700 psia and ISTS SR 3.5.1.5 requires power to be removed from the isolation valve operator ≥ 2000 psia. Per the requirements of CTS 4.0.1 (ITS SR 3.0.1) and CTS 4.0.4 (ITS SR 3.0.4), SRs must be met prior to entry into the Applicability, which in the case of CTS 3.5.1 (ITS LCO 3.5.1) is MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia. Conversely, as stated in the ITS Bases of SR 3.0.1, Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable. Therefore, it is unnecessary to specifically require power to the valve operator be removed in MODE 3 with RCS pressure ≥ 700 psia. Unit 2 CTS 3.5.1 Applicability does not require the SITs to be OPERABLE until in at least MODE 3 with pressurizer pressure ≥ 1750 psia. As such, this additional wording has been removed from Unit 2 ITS SR 3.5.1.5. This changes the CTS by relocating this requirement in CTS 3.5.1 to the Technical Requirements Manual.

The removal of this SIT requirement when in MODE 3 with pressurizer pressure < 1750 psia is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. The ITS retains the requirements for SITs to be OPERABLE when in MODES 1, 2, and 3 with pressurizer pressure ≥ 1750 psia.

When pressurizer pressure is below 1750 psia the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 limit of 2200°F. Therefore, the SITs are not required to be OPERABLE, and the SIT motor operated isolation valves may be closed to isolate the SITs from the RCS during plant shutdown or startup. Also, this change is acceptable because these types of procedural details will continue to be controlled in the TRM. Changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because these SIT requirements are not required by the CTS 3.5.1 and ITS 3.5.1 Applicability, that being MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia. | R1

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Safety Injection Tanks (SITs)

LCO 3.5.1

LCO 3.5.1 ~~[~~Four~~]~~ SITs shall be OPERABLE. 2

3.5.1 Applicability

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure ≥ ~~[700]~~ 1750 psia. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION 3.5.1.a A. One SIT inoperable due to boron concentration not within limits. OR ACTION 3.5.1.a One SIT inoperable due to the inability to verify level or pressure.	A.1 Restore SIT to OPERABLE status.	72 hours
B. One SIT inoperable for reasons other than Condition A.	B.1 Restore SIT to OPERABLE status.	24 hours
ACTION 3.5.1.a. 3.5.1.b C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce pressurizer pressure to < ~~[700]~~ 1750 psia.	6 hours 12 hours 2
ACTION 3.5.1.b D. Two or more SITs inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
LCO 3.5.1.a SR 4.5.1.1.a.2	SR 3.5.1.1	Verify each SIT isolation valve is fully open.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
LCO 3.5.1.b SR 4.5.1.1.a.1	SR 3.5.1.2	Verify borated water volume in each SIT is ≥ ~~[28% narrow range (1802~~ cubic feet~~)~~ and ≤ ~~72% narrow range (1914~~ cubic feet~~)~~~~]~~. 1170 1090	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
LCO 3.5.1.d SR 4.5.1.1.a.1	SR 3.5.1.3	Verify nitrogen cover pressure in each SIT is ≥ ~~[615]~~ psig and ≤ ~~[655]~~ psig. 230 280	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

2

2

2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
LCO 3.5.1.c SR 4.5.1.1.b DOC M01	SR 3.5.1.4	Verify boron concentration in each SIT is ≥ ~~[1500]~~ ppm and ≤ ~~[2800]~~ ppm. (1900, 2200)	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ AND --------NOTE-------- Only required to be performed for affected SIT ---------- Once within 6 hours after each solution volume increase of ≥ ~~[~~1~~]~~% of tank volume that is not the result of addition from the refueling water tank	2

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Safety Injection Tanks (SITs)

LCO 3.5.1

LCO 3.5.1 ~~[~~Four~~]~~ SITs shall be OPERABLE. 2

3.5.1 Applicability

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure ≥ ~~[700]~~ 1750 psia. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION 3.5.1.a A. One SIT inoperable due to boron concentration not within limits. OR ACTION 3.5.1.a One SIT inoperable due to the inability to verify level or pressure.	A.1 Restore SIT to OPERABLE status.	72 hours
B. One SIT inoperable for reasons other than Condition A.	B.1 Restore SIT to OPERABLE status.	24 hours
ACTION 3.5.1.a. 3.5.1.b C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce pressurizer pressure to < ~~[700]~~ 1750 psia.	6 hours 12 hours 2
ACTION 3.5.1.b D. Two or more SITs inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
LCO 3.5.1.a SR 4.5.1.1.a.2	SR 3.5.1.1	Verify each SIT isolation valve is fully open.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
LCO 3.5.1.b SR 4.5.1.1.a.1	SR 3.5.1.2	Verify borated water volume in each SIT is ≥ ~~[28% narrow range (1802~~ cubic feet~~)~~ and ≤ ~~72% narrow range (1914~~ cubic feet~~)~~~~]~~. (1556; 1420)	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
LCO 3.5.1.d SR 4.5.1.1.a.1	SR 3.5.1.3	Verify nitrogen cover pressure in each SIT is ≥ ~~[615]~~ psig and ≤ ~~[655]~~ psig. (500; 650)	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

2

2

2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
LCO 3.5.1.c SR 4.5.1.1.b	SR 3.5.1.4	Verify boron concentration in each SIT is ≥ ~~[1500]~~ 1900 ppm and ≤ ~~[2800]~~ 2200 ppm.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ AND --------NOTE-------- Only required to be performed for affected SIT ---------- Once within 6 hours after each solution volume increase of ≥ ~~[~~1~~]~~% of tank volume that is not the result of addition from the refueling water tank	2 2
SR 4.5.1.1.c	SR 3.5.1.5	Verify power is removed from each SIT isolation valve operator ~~when pressurizer pressure is ≥ [2000] psia~~.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.1, SAFETY INJECTION TANKS (SITs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The purpose ISTS 3.5.1.5 is to ensure power is removed from each SIT isolation valve operator when RCS pressure is above a specified value to further ensure that an active failure could not result in the undetected closure of an SIT motor operated isolation valve. This pressure value is intended to be above the pressure the SITs are required to be OPERABLE. However, the CTS Applicability is MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia and the equivalent Surveillances (Unit 1 CTS 4.5.1.c and Unit 2 CTS 4.5.1.1.c) require power to be removed to the isolation valve operators above 1750 psia or less. Per the requirements of ISTS SR 3.0.1 and ISTS SR 3.0.4, SRs must be met prior to entry into the Applicability, which in the case of ITS LCO 3.5.1 is MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia. Conversely, as stated in the ITS Bases of SR 3.0.1, Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable. Therefore, it is unnecessary to specifically require power to the valve operator be removed "when in MODE 3 with pressure ≥ 1750 psia" and this additional wording has been removed.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Safety Injection Tanks (SITs)

BASES

BACKGROUND

The functions of the ~~[~~four~~]~~ SITs are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA. 2

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

The refill phase of a LOCA follows immediately where reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of the SITs' inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

The SITs are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The SITs are passive components, since no operator or control action is required for them to perform their function. Internal tank pressure is sufficient to discharge the contents to the RCS, if RCS pressure decreases below the SIT pressure.

Each SIT is piped into one RCS cold leg via the injection lines utilized by the High Pressure Safety Injection and Low Pressure Safety Injection (HPSI and LPSI) systems. Each SIT is isolated from the RCS by a motor operated isolation valve and ~~two~~ check valve~~s in series~~. The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident. 1 | R1

The SIT gas and water volumes, gas pressure, and outlet pipe size are selected to allow three of the four SITs to partially recover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three SITs are adequate for this function is consistent with the LOCA assumption that the entire contents of one SIT will be lost via the break during the blowdown phase of a LOCA.

BASES

APPLICABLE SAFETY ANALYSES

The SITs are taken credit for in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the SITs. Reference to the analyses for these DBAs is used to assess changes to the SITs as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of SI flow. These assumptions include signal generation time, equipment starting times, and delivery time due to system piping. In the early stages of a LOCA with a loss of offsite power, the SITs provide the sole source of makeup water to the RCS. (The assumption of a loss of offsite power is required by regulations.) This is because the LPSI pumps, HPSI pumps, and charging pumps cannot deliver flow until the diesel generators (DGs) start, come to rated speed, and go through their timed loading sequence. In cold leg breaks, the entire contents of one SIT are assumed to be lost through the break during the blowdown and reflood phases.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

During this event, the SITs discharge to the RCS as soon as RCS pressure decreases to below SIT pressure. As a conservative estimate, no credit is taken for SI pump flow until the SITs are empty. This results in a minimum effective delay of over 60 seconds, during which the SITs must provide the core cooling function. The actual delay time does not exceed 30 seconds. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA also assumes a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the SITs, with pumped flow then providing continued cooling. As break size decreases, the SITs and HPSI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the SITs continues to decrease until they are not required, and the HPSI pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria, established by 10 CFR 50.46 (Ref. 2) for the ECCS, will be met following a LOCA:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. Maximum fuel element cladding temperature is ≤ 2200°F,

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react, and

d. The core is maintained in a coolable geometry.

Since the SITs discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

Since the SITs are passive components, single active failures are not applicable to their operation. The SIT isolation valves, however, are not single failure proof; therefore, whenever the valves are open, power is removed from their operators and the switch is key locked open.

These precautions ensure that the SITs are available during an accident (Ref. 3). With power supplied to the valves, a single active failure could result in a valve closure, which would render one SIT unavailable for injection. If a second SIT is lost through the break, only two SITs would reach the core. Since the only active failure that could affect the SITs would be the closure of a motor operated outlet valve, the requirement to remove power from these eliminates this failure mode.

The minimum volume requirement for the SITs ensures that three SITs can provide adequate inventory to reflood the core and downcomer following a LOCA. The downcomer then remains flooded until the HPSI and LPSI systems start to deliver flow.

The maximum volume limit is based on maintaining an adequate gas volume to ensure proper injection and the ability of the SITs to fully discharge, as well as limiting the maximum amount of boron inventory in the SITs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

A minimum of ~~25% narrow range level, corresponding to [1790]~~ 1090 cubic feet of borated water, and a maximum of ~~75% narrow range level, corresponding to [1927]~~ cubic feet of borated water, are used in the safety analyses as the volume in the SITs. ~~To allow for instrument inaccuracy, a [28]% narrow range (corresponding to [1802] cubic feet) and a [72]% narrow range (corresponding to [1914] cubic feet) are specified. The analyses are based upon the cubic feet requirements; the percentage figures are provided for operator use because the level indicator provided in the control room is marked in percentages, not in cubic feet.~~ 1170 2 3 2 2 3

The minimum nitrogen cover pressure requirement ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analyses.

The maximum nitrogen cover pressure limit ensures that excessive amounts of gas will not be injected into the RCS after the SITs have emptied.

A minimum pressure of ~~[593]~~ 230 psig and a maximum pressure of ~~[632]~~ 280 psig are used in the analyses. ~~To allow for instrument accuracy, a [615] psig minimum and [655] psig maximum are specified.~~ The maximum allowable boron concentration of ~~[2800]~~ 2200 ppm is based upon boron precipitation limits in the core following a LOCA. Establishing a maximum limit for boron is necessary since the time at which boron precipitation would occur in the core following a LOCA is a function of break location, break size, the amount of boron injected into the core, and the point of ECCS injection. Post LOCA emergency procedures directing the operator to establish simultaneous hot and cold leg injection are based on the worst case minimum boron precipitation time. Maintaining the maximum SIT boron concentration within the upper limit ensures that the SITs do not invalidate this calculation. An excessive boron concentration in any of the borated water sources used for injection during a LOCA could result in boron precipitation earlier than predicted. 4 2

The minimum boron requirements of ~~[1500]~~ 1900 ppm are based on beginning of life reactivity values and are selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. During a large break LOCA, all control element assemblies (CEAs) are assumed not to insert into the core, and the initial reactor shutdown is 2

BASES

APPLICABLE SAFETY ANALYSES (continued)

accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the SITs to prevent a return to criticality during reflood. ~~Although this requirement is similar to the basis for the minimum boron concentration of the refueling water tank (RWT), the minimum SIT concentration is lower than that of the RWT since the SITs need not account for dilution by the RCS.~~ 1

The SITs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the SITs are available to accomplish their core cooling safety function following a LOCA. ~~[~~Four~~]~~ SITs are required to be OPERABLE to ensure that 100% of the contents of ~~[~~three~~]~~ of the SITs will reach the core during a LOCA. 2

This is consistent with the assumption that the contents of one tank spill through the break. If the contents of fewer than three tanks are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an SIT to be considered OPERABLE, the isolation valve must be fully open, power removed above ~~[2000]~~ ~~psig~~, and the limits established in the SR for contained volume, boron concentration, and nitrogen cover pressure must be met. 2

1750 psia

APPLICABILITY

In MODES 1 and 2, and MODE 3 with RCS pressure ≥ ~~700~~ psia, the SIT OPERABILITY requirements are based on an assumption of full power operation. Although cooling requirements decrease as power decreases, the SITs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist. 2

1750

This LCO is only applicable at pressures ≥ ~~700~~ psia. Below ~~700~~ psia, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F. 2

1750

1750

In MODE 3, at pressures < ~~700~~ psia, and in MODES 4, 5, and 6, the SIT motor operated isolation valves are closed to isolate the SITs from the RCS. ~~This allows RCS cooldown and depressurization without discharging the SITs into the RCS or requiring depressurization of the SITs.~~ 5

during unit shutdown and startup

BASES

ACTIONS

A.1

If the boron concentration of one SIT is not within limits, it must be returned to within the limits within 72 hours. In this condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced, but the reduced concentration effects on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the SIT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of three SITs, the consequences are less severe than they would be if an SIT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

The combination of redundant level and pressure instrumentation for any single SIT provides sufficient information so that it is not worthwhile to always attempt to correct drift associated with one instrument, with the resulting radiation exposures during entry into containment, as there is sufficient time to repair one in the event that a second one became inoperable. Because these instruments do not initiate a safety action, it is reasonable to extend the allowable outage time for them. While technically inoperable, the SIT will be available to fulfill its safety function during this time and, thus, this Completion Time results in a negligible increase in risk.

B.1

If one SIT is inoperable, for reasons other than boron concentration or the inability to verify level or pressure, the SIT must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three SITs cannot be assumed to reach the core during a LOCA as is assumed in Appendix K (Ref. 5).

CE NPSD-994 (Ref. 6) provides a series of deterministic and probabilistic findings that support the 24 hour Completion Time as having no affect on risk as compared to shorter periods for restoring the SIT to OPERABLE status.

BASES

ACTIONS (continued)

C.1 and C.2

If the SIT cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < ~~700~~ 1750 psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. 2

D.1

If more than one SIT is inoperable, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification that each SIT isolation valve is fully open, before power is removed from its valve operator, as indicated in the control room, ensures that SITs are available for injection and ensures timely discovery if a valve should be partially closed. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve should not change position with power removed, a closed valve could result in not meeting accident analysis assumptions. ~~[A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure the unlikelihood of a mispositioned isolation valve.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.2 and SR 3.5.1.3

SIT borated water volume and nitrogen cover pressure should be verified to be within specified limits in order to ensure adequate injection during a LOCA. [Due to the static design of the SITs, a 12 hour Frequency usually allows the operator sufficient time to identify changes before the limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends. 2

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement. 2
----------]

SR 3.5.1.4

Thirty-one days is reasonable for verification to determine that each SIT's boron concentration is within the required limits, because the static design of the SITs limits the ways in which the concentration can be changed. [The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. 2
Sampling the affected SIT within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water is from the RWT, because the water contained in the RWT is within the SIT boron concentration requirements. This is consistent with the recommendations of NUREG-1366 (Ref. 4). 6

the boron concentration of

and assumes the RWT has not been diluted since the last time the RWT boron concentration was verified to be within limits.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.5.1.5

Verification that power is removed from each SIT isolation valve operator ~~when the pressurizer pressure is ≥ 2000 psia~~ ensures that an active failure could not result in the undetected closure of an SIT motor operated isolation valve. If this were to occur, only two SITs would be available for injection, given a single failure coincident with a LOCA~~. [Since installation and~~ removal of power to the SIT isolation valve operators is conducted under administrative control~~, the 31 day Frequency was chosen to provide additional assurance that power is removed~~.

2

R

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is < 2000 psia, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startups or shutdowns.~~

1

BASES

REFERENCES

1. FSAR, ~~Section [6.3]~~.

2. 10 CFR 50.46.

3. FSAR, ~~Chapter [15]~~.

4. Draft NUREG-1366, February 1990.

5. 10 CFR 50 Appendix K.

6. CE NPSD-994, "CEOG Joint Applications Report for Safety Injection Tank AOT/STI Extension," May 1995.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Safety Injection Tanks (SITs)

BASES

BACKGROUND

The functions of the ~~[~~four~~]~~ SITs are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA. 2

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

The refill phase of a LOCA follows immediately where reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of the SITs' inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

The SITs are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The SITs are passive components, since no operator or control action is required for them to perform their function. Internal tank pressure is sufficient to discharge the contents to the RCS, if RCS pressure decreases below the SIT pressure.

Each SIT is piped into one RCS cold leg via the injection lines utilized by the High Pressure Safety Injection and Low Pressure Safety Injection (HPSI and LPSI) systems. Each SIT is isolated from the RCS by a motor operated isolation valve and ~~two~~ check valve~~s in series~~. The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident. 1 | R1

The SIT gas and water volumes, gas pressure, and outlet pipe size are selected to allow three of the four SITs to partially recover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three SITs are adequate for this function is consistent with the LOCA assumption that the entire contents of one SIT will be lost via the break during the blowdown phase of a LOCA.

BASES

APPLICABLE SAFETY ANALYSES

The SITs are taken credit for in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the SITs. Reference to the analyses for these DBAs is used to assess changes to the SITs as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of SI flow. These assumptions include signal generation time, equipment starting times, and delivery time due to system piping. In the early stages of a LOCA with a loss of offsite power, the SITs provide the sole source of makeup water to the RCS. (The assumption of a loss of offsite power is required by regulations.) This is because the LPSI pumps, HPSI pumps, and charging pumps cannot deliver flow until the diesel generators (DGs) start, come to rated speed, and go through their timed loading sequence. In cold leg breaks, the entire contents of one SIT are assumed to be lost through the break during the blowdown and reflood phases.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

During this event, the SITs discharge to the RCS as soon as RCS pressure decreases to below SIT pressure. As a conservative estimate, no credit is taken for SI pump flow until the SITs are empty. This results in a minimum effective delay of over 60 seconds, during which the SITs must provide the core cooling function. The actual delay time does not exceed 30 seconds. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA also assumes a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the SITs, with pumped flow then providing continued cooling. As break size decreases, the SITs and HPSI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the SITs continues to decrease until they are not required, and the HPSI pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria, established by 10 CFR 50.46 (Ref. 2) for the ECCS, will be met following a LOCA:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. Maximum fuel element cladding temperature is ≤ 2200°F,

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react, and

d. The core is maintained in a coolable geometry.

Since the SITs discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

Since the SITs are passive components, single active failures are not applicable to their operation. The SIT isolation valves, however, are not single failure proof; therefore, whenever the valves are open, power is removed from their operators and the switch is key locked open.

These precautions ensure that the SITs are available during an accident (Ref. 3). With power supplied to the valves, a single active failure could result in a valve closure, which would render one SIT unavailable for injection. If a second SIT is lost through the break, only two SITs would reach the core. Since the only active failure that could affect the SITs would be the closure of a motor operated outlet valve, the requirement to remove power from these eliminates this failure mode.

The minimum volume requirement for the SITs ensures that three SITs can provide adequate inventory to reflood the core and downcomer following a LOCA. The downcomer then remains flooded until the HPSI and LPSI systems start to deliver flow.

The maximum volume limit is based on maintaining an adequate gas volume to ensure proper injection and the ability of the SITs to fully discharge, as well as limiting the maximum amount of boron inventory in the SITs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

A minimum of ~~25% narrow range level, corresponding to [1790]~~ 1420 cubic feet of borated water, and a maximum of ~~75% narrow range level, corresponding to [1927]~~ cubic feet of borated water, are used in the safety analyses as the volume in the SITs. ~~To allow for instrument inaccuracy, a [28]% narrow range (corresponding to [1802] cubic feet) and a [72]% narrow range (corresponding to [1914] cubic feet) are specified. The analyses are based upon the cubic feet requirements; the percentage figures are provided for operator use because the level indicator provided in the control room is marked in percentages, not in cubic feet.~~ 1556 2 3 2 2 3

The minimum nitrogen cover pressure requirement ensures that the contained gas volume will generate discharge flow rates during injection that are consistent with those assumed in the safety analyses.

The maximum nitrogen cover pressure limit ensures that excessive amounts of gas will not be injected into the RCS after the SITs have emptied.

A minimum pressure of ~~[593]~~ 500 psig and a maximum pressure of ~~[632]~~ 650 psig are used in the analyses. ~~To allow for instrument accuracy, a [615] psig minimum and [655] psig maximum are specified.~~ 4 The maximum allowable boron concentration of ~~[2800]~~ 2200 ppm is based upon boron precipitation limits in the core following a LOCA. 2 Establishing a maximum limit for boron is necessary since the time at which boron precipitation would occur in the core following a LOCA is a function of break location, break size, the amount of boron injected into the core, and the point of ECCS injection. Post LOCA emergency procedures directing the operator to establish simultaneous hot and cold leg injection are based on the worst case minimum boron precipitation time. Maintaining the maximum SIT boron concentration within the upper limit ensures that the SITs do not invalidate this calculation. An excessive boron concentration in any of the borated water sources used for injection during a LOCA could result in boron precipitation earlier than predicted.

The minimum boron requirements of ~~[1500]~~ 1900 ppm are based on beginning of life reactivity values and are selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. 2 During a large break LOCA, all control element assemblies (CEAs) are assumed not to insert into the core, and the initial reactor shutdown is

BASES

APPLICABLE SAFETY ANALYSES (continued)

accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the SITs to prevent a return to criticality during reflood. ~~Although this requirement is similar to the basis for the minimum boron concentration of the refueling water tank (RWT), the minimum SIT concentration is lower than that of the RWT since the SITs need not account for dilution by the RCS.~~ 1

The SITs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the SITs are available to accomplish their core cooling safety function following a LOCA. ~~[~~Four~~]~~ SITs are required to be OPERABLE to ensure that 100% of the contents of ~~[~~three~~]~~ of the SITs will reach the core during a LOCA. 2

This is consistent with the assumption that the contents of one tank spill through the break. If the contents of fewer than three tanks are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an SIT to be considered OPERABLE, the isolation valve must be fully open, power removed above ~~[2000]~~ ~~psig~~, and the limits established in the SR for contained volume, boron concentration, and nitrogen cover pressure must be met. 2

1750 psia

APPLICABILITY

In MODES 1 and 2, and MODE 3 with RCS pressure ≥ ~~700~~ psia, the SIT OPERABILITY requirements are based on an assumption of full power operation. Although cooling requirements decrease as power decreases, the SITs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist. 2

1750

This LCO is only applicable at pressures ≥ ~~700~~ psia. Below ~~700~~ psia, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F. 2

1750 1750

In MODE 3, at pressures < ~~700~~ psia, and in MODES 4, 5, and 6, the SIT motor operated isolation valves are closed to isolate the SITs from the RCS. ~~This allows RCS cooldown and depressurization without discharging the SITs into the RCS or requiring depressurization of the SITs.~~ 5

1750

during unit shutdown and startup

BASES

ACTIONS

A.1

If the boron concentration of one SIT is not within limits, it must be returned to within the limits within 72 hours. In this condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced, but the reduced concentration effects on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the SIT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of three SITs, the consequences are less severe than they would be if an SIT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

The combination of redundant level and pressure instrumentation for any single SIT provides sufficient information so that it is not worthwhile to always attempt to correct drift associated with one instrument, with the resulting radiation exposures during entry into containment, as there is sufficient time to repair one in the event that a second one became inoperable. Because these instruments do not initiate a safety action, it is reasonable to extend the allowable outage time for them. While technically inoperable, the SIT will be available to fulfill its safety function during this time and, thus, this Completion Time results in a negligible increase in risk.

B.1

If one SIT is inoperable, for reasons other than boron concentration or the inability to verify level or pressure, the SIT must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three SITs cannot be assumed to reach the core during a LOCA as is assumed in Appendix K (Ref. 5).

CE NPSD-994 (Ref. 6) provides a series of deterministic and probabilistic findings that support the 24 hour Completion Time as having no affect on risk as compared to shorter periods for restoring the SIT to OPERABLE status.

BASES

ACTIONS (continued)

C.1 and C.2

If the SIT cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < ~~700~~ 1750 psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. 2

D.1

If more than one SIT is inoperable, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification that each SIT isolation valve is fully open, as indicated in the control room before power is removed from its valve operator, ensures that SITs are available for injection and ensures timely discovery if a valve should be partially closed. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve should not change position with power removed, a closed valve could result in not meeting accident analysis assumptions. ~~[A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure the unlikelihood of a mispositioned isolation valve.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.2 and SR 3.5.1.3

SIT borated water volume and nitrogen cover pressure should be verified to be within specified limits in order to ensure adequate injection during a LOCA. [Due to the static design of the SITs, a 12 hour Frequency usually allows the operator sufficient time to identify changes before the limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends. 2

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement. 2
----------]

SR 3.5.1.4

Thirty-one days is reasonable for verification to determine that each SIT's boron concentration is within the required limits, because the static design of the SITs limits the ways in which the concentration can be changed. [The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. 2
Sampling the affected SIT within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water is from the RWT, because the water contained in the RWT is within the SIT boron concentration requirements. This is consistent with the recommendations of NUREG-1366 (Ref. 4). 6

the boron concentration of

and assumes the RWT has not been diluted since the last time the RWT boron concentration was verified to be within limits.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.5.1.5

Verification that power is removed from each SIT isolation valve operator ~~when the pressurizer pressure is ≥ 2000 psia~~ ensures that an active failure could not result in the undetected closure of an SIT motor operated isolation valve. If this were to occur, only two SITs would be available for injection, given a single failure coincident with a LOCA~~. [Since installation and~~ removal of power to the SIT isolation valve operators is conducted under administrative control~~, the 31 day Frequency was chosen to provide additional assurance that power is removed~~.

2

R

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is < 2000 psia, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startups or shutdowns.~~

1

BASES

REFERENCES

1. FSAR, ~~Section [6.3]~~.

2. 10 CFR 50.46.

3. FSAR, ~~Chapter [15]~~.

4. Draft NUREG-1366, February 1990.

5. 10 CFR 50 Appendix K.

6. CE NPSD-994, "CEOG Joint Applications Report for Safety Injection Tank AOT/STI Extension," May 1995.

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.1, BASES, SAFETY INJECTION TANKS (SITs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The SIT instrument range percentage values, with and without allowance for instrument inaccuracy, that correspond to the minimum and maximum SIT volumes in cubic feet of borated water are deleted. The analyses are based upon the cubic feet requirements; the percentage values are provided for operator use. Plant procedures implement ITS SR 3.5.1.2 to verify borated water volume in each SIT and include appropriate unit of measurement and instrument accuracy allowances. Therefore, the instrument values and associated description for operator use are unnecessary. The SIT volume values retained in ITS 3.5.1.2 reflect the SIT volume values assumed in the PSL safety analysis.

4. The SIT instrument range values with allowance for instrument inaccuracy, correspond to the minimum and maximum SIT nitrogen pressures are deleted. The analyses are based upon the actual pressures; the instrument accuracy values are provided for operator use. Plant procedures implement ITS SR 3.51.3 to verify nitrogen cover pressure in each SIT and include appropriate unit of measurement and instrument accuracy allowances. Therefore, the instrument accuracy range in the ITS Bases is unnecessary. The SIT nitrogen cover pressure values retained in ITS 3.5.1.3 reflect the SIT pressure values assumed in the PSL safety analysis.

5. The ISTS Bases statement regarding why an SIT is isolated when not within the ITS Applicability is changed to state that the SIT isolation valves are closed to isolate the SITs from the RCS during unit shutdown and startup. Plant procedures implement shutdown and startup requirements including when the SITs may and may not be isolated from the RCS. This change is consistent with the ITS SR 3.5.1.5 description.

6. Additional clarification is added from the CTS to the Bases of ITS 3.5.1.4 to state that the SIT boron concentration verification is unnecessary because the boron concentration of the water contained in the RWT is within the SIT boron concentration requirements. This assumes the RWT has not been diluted since the last time the RWT boron concentration was verified to be within limits.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.5.1, SAFETY INJECTION TANKS (SITs)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

3.5.2 ECCS - Operating

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.5.2

3.5.2 Two ~~independent~~ ECCS ~~subsystems~~ trains shall be OPERABLE ~~with each subsystem comprised of:~~

~~a. One OPERABLE high-pressure safety injection (HPSI) pump,~~

~~b. One OPERABLE low-pressure safety injection pump,~~

~~c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and~~

LA01

~~**NOTE**~~
~~One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e.~~

LA02

~~d. One OPERABLE charging pump.~~

LA01

Applicability

APPLICABILITY: MODES 1, 2 and 3 with pressurizer pressure ≥ 1750 psia.

ACTION:

Condition A
Required Action A.1

Required Action C .1, C.2

~~a.~~ ~~1.~~ ~~With~~ one ~~ECCS~~ subsystem inoperable ~~only because its associated~~ LPSI ~~train is inoperable~~, restore ~~the inoperable~~ subsystem to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in ~~at least~~ ~~HOT STANDBY~~ MODE 3 within ~~the next~~ 6 hours and ~~in HOT SHUTDOWN~~ Reduce pressurizer pressure to < 1750 psia within ~~the following~~ ~~6~~ 12 hours.

A02

Condition B
Required Action B.1

Required Action C .1, C.2

~~2.~~ ~~With~~ one ~~ECCS subsystem~~ or more trains train(s) inoperable for reasons other than condition a~~.1.~~, restore ~~the inoperable subsystem~~ to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least~~ ~~HOT STANDBY~~ MODE 3 within ~~the next~~ 6 hours and ~~in HOT SHUTDOWN~~ Reduce pressurizer pressure to < 1750 psia within ~~the following~~ ~~6~~ 12 hours.

L01

A02

~~b.~~ ~~In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.~~

L02

Add proposed Action D

L01

A01

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 ~~Each ECCS subsystem shall be demonstrated OPERABLE:~~

SR 3.5.2.1

~~a.~~ In accordance with the Surveillance Frequency Control Program by verifying ~~that~~ the following valves are in the ~~indicated~~ position~~s~~ with power to the valve operator~~s~~ removed:

Valve Number	Valve Function	Valve Position
~~1.~~ V-3659	~~1.~~ Mini-flow isolation	~~1.~~ Open
~~2.~~ V-3660	~~2.~~ Mini-flow isolation	~~2.~~ Open

SR 3.5.2.2, SR 3.5.2.3

~~b.~~ In accordance with the Surveillance Frequency Control Program by:

ECCS

SR 3.5.2.2

~~1.~~ Verify~~ing that~~ each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.*

SR 3.5.2.3

~~2.~~ Verify~~ing~~ ECCS ~~train~~ locations susceptible to gas accumulation are sufficiently filled with water.

.

~~c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:~~

~~1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and~~

~~2. At least once daily of the areas affected within containment by the containment entry and during the final entry when CONTAINMENT INTEGRITY is established.~~

LA03

~~d.~~ In accordance with the Surveillance Frequency Control Program by:

1. Verifying proper operation of the open permissive interlock (OPI) and the valve open/high SDCS pressure alarms for isolation valves V3651, V3652, V3480, V3481.

See ITS 3.4.14

each ECCS train

SR 3.5.2.9

~~2.~~ ~~A~~ visual inspection ~~of the~~ containment sump ~~and verifying that the subsystem~~ suction inlet~~s~~ are not restricted by debris and ~~that the sump components~~ (trash racks~~,~~ screens~~, etc.)~~ show no evidence of structural distress or corrosion.

Verify by

the suction inlet

and

abnormal

SR 3.5.2.2 Note * Not required to be met for system vent flow paths opened under administrative control.

A01

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.6, SR 3.5.2.7, SR 3.5.2.8

~~e.~~ In accordance with the Surveillance Frequency Control Program~~, during shutdown, by:~~

L03

L04

ECCS

that is not locked, sealed, or otherwise secured in position,

SR 3.5.2.6

~~1.~~ Verify~~ing that~~ each automatic valve in the flow path~~s~~ actuates to its correct position on ~~a Safety Injection~~ Actuation Signal.

an actual or simulated

L05 LA01

ECCS

s

SR 3.5.2.7

~~2.~~ Verify~~ing that~~ each ~~of the following~~ pump~~s~~ start automatically ~~upon receipt of a Safety Injection~~ Actuation Signal;

on an actual or simulated

L05

~~a. High-Pressure Safety Injection Pumps.~~

~~b. Low-Pressure Safety Injection Pumps.~~

~~c. Charging Pumps.~~

LA01

SR 3.5.2.8

~~3.~~ Verify~~ing that~~ upon receipt of an actual or simulated ~~Recirculation~~ Actuation Signal: each low-pressure safety injection pump stops, ~~each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes.~~

LA04

ECCS

SR 3.5.2.4

~~f.~~ ~~By~~ verify~~ing that~~ each ~~of the following~~ pump~~s~~ ~~develops the specified total~~ developed head ~~when tested pursuant to the~~ INSERVICE TESTING PROGRAM.

at the test flow point is greater than or equal to the required developed head in accordance with the

~~1. High-Pressure Safety Injection pumps.~~

~~2. Low-Pressure Safety Injection pumps.~~

LA01

SR 3.5.2.5

Add proposed SR 3.5.2.5

M01

SR 3.5.2.10

Add proposed SR 3.5.2.10

Add proposed SR 3.5.2.11

M02

A01

~~**EMERGENCY CORE COOLLING SYSTEMS**~~

~~**SURVEILLANCE REQUIREMENTS** (Continued)~~

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A01

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.5.2

3.5.2 Two ~~independent~~ Emergency Core Cooling System (ECCS) ~~subsystems~~ trains shall be OPERABLE ~~with each subsystem comprised of:~~

~~a. One OPERABLE high pressure safety injection pump,~~

~~b. One OPERABLE low pressure safety injection pump, and~~

~~c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and~~

LA01

~~**NOTE**~~

~~One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e.~~

LA02

~~d. One OPERABLE charging pump.~~

LA01

Applicability **APPLICABILITY:** MODES 1, 2, and 3 with pressurizer pressure ≥ 1750 psia.

ACTION:

Condition A
Required Action A.1

~~a. 1. With~~ one ~~ECCS~~ subsystem inoperable ~~only because its associated train is inoperable~~, restore ~~the inoperable~~ subsystem to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ 6 hours.

LPSI

12

MODE 3

or more trains

train(s)

Reduce pressurizer pressure to < 1750 psia

A02

L01

Condition B
Required Action B.1

~~2. With~~ one ~~ECCS subsystem~~ inoperable for reasons other than condition a~~.1.~~, restore ~~the inoperable subsystem~~ to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~in HOT SHUTDOWN~~ within ~~the following~~ 6 hours.

Required Action C .1, C.2

12

MODE 3

Reduce pressurizer pressure to < 1750 psia

A02

~~b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~

L02

Add proposed Action D

L01

A01

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. V3733 V3734	a. SIT Vent Valves	a. Locked Closed
b. V3735 V3736	b. SIT Vent Valves	b. Locked Closed
c. V3737 V3738 V3739 V3740	c. SIT Vent Valves	c. Locked Closed

L07

SR 3.5.2.2 b. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.*

SR 3.5.2.3 c. In accordance with the Surveillance Frequency Control Program by verifying ECCS locations susceptible to gas accumulation are sufficiently filled with water.

. d. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. At least once daily of the areas affected within containment by the containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

LA03

e. In accordance with the Surveillance Frequency Control Program by:

1. Verifying automatic isolation and interlock action of the shutdown cooling system from Reactor Coolant System when RCS pressure (actual or simulated) is greater than or equal to 515 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when RCS pressure (actual or simulated) is greater than or equal to 276 psia.

See ITS 3.4.14

SR 3.5.2.2 Note * Not required to be met for system vent flow paths opened under administrative control.

A01

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

each ECCS train

SR 3.5.2.1

2. ~~A~~ visual inspection ~~of the~~ containment sump ~~and verifying that the subsystem~~ suction inlet~~s~~ are not restricted by debris and ~~that the sump components (~~trash racks~~,~~ screens~~, etc.)~~ show no evidence of structural distress or corrosion.

Verify by

and

abnormal

3. Verifying that a minimum total of 173 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 70.5 ± 0.5 grams of TSP from a TSP storage basket is submerged, without agitation, in 10.0 ± 0.1 gallons of 120 ± 10°F borated water representative of the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.

See ITS 3.5.5

SR 3.5.2.6, SR 3.5.2.7, SR 3.5.2.8

~~f.~~ In accordance with the Surveillance Frequency Control Program~~, during shutdown, by~~:

L03

ECCS

that is not locked, sealed, or otherwise secured in position,

L04

SR 3.5.2.6

~~1.~~ Verify~~ing that~~ each automatic valve in the flow path~~s~~ actuates to its correct position on ~~SIAS and/or RAS test~~ signal~~s~~.

an actual or simulated actuation

L05

ECCS

s

SR 3.5.2.7

~~2.~~ Verifying that each ~~of the following~~ pump~~s~~ start automatically ~~upon receipt of a Safety Injection~~ Actuation ~~Test~~ Signal:

on an actual or simulated

L05

~~a. High-Pressure Safety Injection pumps.~~

LA01

~~b. Low-Pressure Safety Injection pumps.~~

~~c. Charging Pumps~~

SR 3.5.2.8

~~3.~~ Verify~~ing that~~ upon receipt of an actual or simulated ~~Recirculation~~ Actuation Signal: each low-pressure safety inject ion pump stops, ~~each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes.~~

LA04

ECCS

SR 3.5.2.4

~~g.~~ ~~By~~ verify~~ing that~~ each ~~of the following~~ pump~~s~~ ~~develops the specified total~~ developed head ~~when tested pursuant to the~~ INSERVICE TESTING PROGRAM:

at the test flow point is greater than or equal to the required developed head in accordance with the

~~1. High-Pressure Safety Injection pumps.~~

LA01

~~2. Low-Pressure Safety Injection pumps.~~

SR 3.5.2.9

~~h.~~ ~~By~~ verify~~ing the~~ correct position ~~of~~ each ~~electrical and/or mechanical~~ position stop for ~~the following~~ ECCS throttle valve~~s~~:

is in the

each

listed below

~~1. During valve stroking operation or following maintenance on the valve and prior to declaring the valve OPERABLE when the ECCS subsystems are required to be OPERABLE~~.

L06

SR 3.5.2.5

Add proposed SR 3.5.2.5

M01

SR 3.5.2.10

Add proposed SR 3.5.2.10

Add proposed SR 3.5.2.11

M02

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.5.2.9

~~2.~~ In accordance with the Surveillance Frequency Control Program.

HPSI System Valve Number		LPSI System Valve Number	
~~a.~~	HCV 3616/3617	~~a.~~	HCV 3615
~~b.~~	HCV 3626/3627	~~b.~~	HCV 3625
~~c.~~	HCV 3636/3637	~~c.~~	HCV 3635
~~d.~~	HCV 3646/3647	~~d.~~	HCV 3645
~~e.~~	V3523/V3540		

DISCUSSION OF CHANGES
ITS 3.5.2, ECCS - OPERATING

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.5.2 Actions a.1 and a.2 require in part, “or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.” when one ECCS subsystem cannot be restored to OPERABLE status within the required time. ITS 3.5.2 Required Action C.2 requires, when the Required Actions and associated Completion Times are not met, reducing pressurizer pressure to < 1750 psia within 12 hours (i.e., within the following 6 hours). This changes the end state in CTS 3.5.2 Actions a.1 and a.2 from HOT SHUTDOWN (MODE 4) to MODE 3 with pressurizer pressure < 1750 psia.

The purpose of CTS 3.5.2 Action a.1 and a.2 is to place the plant in a condition in which the equipment is no longer required. CTS 3.5.2 Action a.1 and a.2 provide appropriate actions to take for an inoperable ECCS subsystem when the unit is in MODE 1, 2, or 3 with pressurizer pressure is ≥ 1750 psia. The ECCS subsystems are not required to be OPERABLE when pressurizer pressure is < 1750 psia, as stated in CTS 3.5.2. Therefore, in accordance with CTS 3.0.2 (ITS LCO 3.0.2), when pressurizer pressure is < 1750 psia, the requirements to be in MODE 4 in CTS 3.5.1 Action a.1 and a.2 are no longer required to be completed. The ECCS subsystems, with pressurizer pressure below 1750 psia, are no longer required to be OPERABLE.

These changes are designated as administrative changes and are acceptable because they maintain the current requirement to place the unit in a condition in which the ECCS subsystems are no longer required consistent with the requirements of CTS 3.0.2 (ITS LCO 3.0.2) and, therefore, do not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.5.2 requires an OPERABLE charging pump. Unit 1 CTS 4.5.2.e.2 and f., and Unit 2 4.5.2.f.2 and g., require performance testing of the ECCS subsystems. CTS does not include a Surveillance Requirement for performance testing of the charging pumps. ITS SR 3.5.2.5 requires verification that the charging pumps develop a minimum required flow. ITS SR 3.5.2.5 requires this SR to be conducted in accordance with the INSERVICE TESTING PROGRAM. This changes the CTS by adding the ITS requirement to verify each charging pump develops a minimum required flow.

The purpose of ITS SR 3.5.2.5 is to verify charging flow within a testing Frequency required in accordance with the INSERVICE TESTING PROGRAM. This change is acceptable because it provides additional assurance that each charging pump will be capable of performing its function. This change is designated as more restrictive because it adds an SR to the CTS.

M02 CTS 4.5.2 does not contain surveillances for boric acid makeup tank (BAMT) volume and the BAMT concentration for the charging pump ECCS function. Similarly, ISTS 3.5.2 does not contain a surveillance for the BAMT volume and the boron concentration for the charging pump ECCS function. The ISTS 3.5.2 assumes the RWT as the ECCS water source for LPSI, HPSI, and charging ECCS subsystems. The PSL design connects the ECCS charging subsystem to the BAMTs. Though the charging subsystem can be aligned to the RWT, and the charging pump suction automatically aligns to the RWT on Volume Control Tank low level, the RWT to charging pump suction valve gets a close signal on a safety injection actuation signal. This changes the CTS by adding Surveillance Requirements for the BAMTs.

The Note to CTS 3.5.2.d refers to CTS 3.1.2.2 for charging subsystem flow path requirements and CTS 3.1.2.2 refers to CTS 3.1.2.8 for BAMT requirements including volume and boron concentration, respectively. However, these boration requirements are not related to the ECCS OPERABILITY but, rather, provide boration requirements to ensure the capability to restore shutdown margin during normal operations. The boration related Specifications, including CTS 3.1.2.2 and 3.1.2.8, are relocated to a licensee controlled document (e.g, Technical Requirements Manual). See DOC LA02 for a discussion of changes related to the relocation of the CTS 3.1.2.2 and 3.1.2.8 references in CTS 3.5.2.d Note.

The new ITS SR 3.5.2.10 and ITS SR 3.5.2.11 require verification that each BAMT volume is greater than or equal to the specified minimum required volume, and that each BAMT boron concentration is greater than or equal to the specified minimum required boron concentration. These Surveillances are necessary to periodically verify that a sufficient usable BAMT water volume level and boron concentration are available to the charging pump for injection during a small break LOCA (SBLOCA). PSL performed an engineering evaluation for the SBLOCA Analysis and determined that a minimum BAMT water volume of 3675 gallons for each tank provides adequate volume to meet the charging pump flow requirement credited in the SBLOCA analysis. A minimum BAMT water volume of 4000 gallons was selected to provide additional margin for future variations in the analysis. Additionally, although boron concentration from the charging flow is not directly credited in the SBLOCA analysis to maintain subcriticality, a minimum boron concentration that is equivalent to the minimum RWT boron concentration (i.e., ≥ 1900 ppm) is used for the charging pump flow.

The minimum boron concentration in the BAMT is important for the post-LOCA return to criticality analysis. This analysis determines the minimum sump boron concentration at the time of simultaneous hot/cold leg injection to confirm return to criticality does not occur. This analysis uses the minimum boron from the available sources, which includes the BAMTs. Currently, the analysis is strictly based on the RCS boron and minimum RWT boron concentration. With the

BAMT boron requirement being added to ITS 3.5.2, ECCS -Operating, the minimum BAMT boron concentration must be at least equal to the RWT minimum boron concentration. The maximum BAMT boron concentration is used in the post-LOCA long term boron precipitation analysis. The current analysis considers the volume of two full BAMTs (9975 gallons/tank) at a boron concentration that bounds the current maximum specified BAMT boron concentration parameters (volume and boron concentration). Thus, there is no need to specify a maximum boron concentration in ITS 3.5.2, ECCS – Operating.

PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, SR 3.5.2.10 and SR 3.5.2.11 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 7 days consistent with the ISTS SR 3.5.4.2 Frequency for the RWT, and consistent with the CTS 4.1.2.8.a Frequency for the BAMT volume and boron concentration. The initial Frequency considers that the BAMT volume and boron concentration are normally stable. This changes the CTS by adding a Surveillance Requirement to verify that each BAMT volume is greater than or equal to the specified minimum required volume in accordance with the Surveillance Frequency Control Program and by adding a Surveillance Requirement to verify that each BAMT boron concentration is greater than or equal to the specified minimum required boron concentration in accordance with the Surveillance Frequency Control Program.

The purpose of this Surveillance is to assure that the necessary quality of systems and components is maintained. The Frequency is proposed to be specified in accordance with the Surveillance Frequency Control Program and is acceptable because the periodic Frequency is placed under licensee control pursuant to the methodology described in NEI 04-10, which is specified in ITS Section 5.5. The SFCP provides the necessary administrative controls to require that surveillances related to testing, calibration and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Florida Power & Light Company (FPL) has determined that the control of the periodic Frequency in accordance with the SFCP is consistent with the intent of TSTF-425, Revision 3, and with the NRC's model SE dated July 6, 2009 (*74 FR 32001*). The proposed Frequency of SR 3.5.2.10 is a periodic frequency and does not meet the scope exclusion criteria identified in Section 1.0, "Introduction," of the TSTF-425 model SE.

The SFCP was established as described in FPL (PSL Unit 1 and Unit 2) “Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program” (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066). This change is designated as more restrictive because it adds additional requirements to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.5.2 states, in part, that two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of: (a) one OPERABLE high pressure safety injection (HPSI) pump, (b) one OPERABLE low pressure safety injection pump, (c) an independent OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal, and (d) one OPERABLE charging pump. Similarly, Unit 1 CTS 4.5.2.e.2 and f, and Unit 2 CTS 4.5.2.f.2 and g, state, in part, the ECCS pumps (i.e., HPSI, LPSI, Charging) to be tested that are part of an OPERABLE ECCS train. ITS 3.5.2 does not contain this level of detail of what constitutes OPERABILITY. This changes the CTS by moving details concerning what constitutes an OPERABLE ECCS train, and that the two systems must be "independent" to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for two ECCS trains to be OPERABLE and is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.5.2.d Note states “One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e”. ITS LCO 3.5.2 does not include details associated with the flow path requirements necessary to support a ECCS charging subsystem. This changes the CTS by removing details related to a ECCS charging subsystem and relocating required details to the ITS Bases.

As stated in the NRC safety evaluation (SE) associated with the Unit 2 extended power uprate (See NRC to FPL letter dated July 9, 2012; NRC ADAMS Accession No. ML12268A167), the purpose of the CTS Note is to assure that ECCS sources credited in the LOCA analysis are available to support the charging subsystems. ITS LCO 3.5.2 states that two ECCS trains shall be OPERABLE. The details of the OPERABILITY requirements associated with the

ECCS trains are addressed by the Surveillance Requirements and the ITS Bases.

The purpose of CTS 3.1.2.2 requirements (referenced in CTS 3.5.2.d Note) is to ensure a means to control the chemical neutron absorber (boron) concentration in the Reactor Coolant System (RCS) is available and to provide adequate requirements to restore shutdown margin during normal operations. These boration requirements are not related to the ECCS OPERABILITY requirements. Referencing reactivity control Specifications for compliance with ECCS is inappropriate and could result in requirements inconsistent with the LOCA analysis. In addition, the boration-related Specifications, including the Specifications referenced in the CTS 3.5.2.d Note are relocated to a licensee controlled document (e.g., Technical Requirements Manual) - see ITS 3.1 Deleted - Relocated Specifications. Therefore, to avoid confusion between boration requirements, as specified in CTS 3.1.2.2, and the charging subsystem requirements to support ECCS OPERABILITY, the ITS Bases relocates sufficient detail to the ITS Bases to assure that ECCS sources (e.g., charging subsystems) credited in the LOCA analysis are available. The proposed ITS Bases states, in part:

> "Charging pump flow paths take suction from OPERABLE BAMTs (e.g., volume within required limit) and supplies the RCS via the normal charging lines.
>
> One charging pump flow path taking suction from either a single BAMT via the associated boric acid makeup pump, or both BAMTs via both boric acid makeup pumps. A second charging pump flow path taking suction from either a single BAMT via gravity feed valves, or both BAMTs via gravity feed valves."

The removal of these details for charging pump flow paths to meet operability requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS LCO 3.5.2 and the associated Surveillance Requirements retain sufficient requirements to assure two OPERABLE ECCS trains, which includes two OPERABLE charging subsystems. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA03 *(Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* Unit 1 CTS 4.5.2.c and Unit 2 4.5.2.d state "By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed: For all accessible areas of the containment prior

to establishing CONTAINMENT INTEGRITY, and at least once daily of the areas affected within containment by the containment entry and during the final entry when CONTAINMENT INTEGRITY is established." ITS 3.5.2 does not include these Surveillances. This changes the CTS by removing the Surveillance for visual inspections inside Containment and relocating them to the Technical Requirements Manual.

The removal of these visual inspections from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. The ITS retains the Surveillance to perform a visual inspection of each ECCS train containment sump suction inlet to ensure that it is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion. Periodic inspections of each ECCS train containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because visual inspections inside Containment are being removed from the Technical Specifications. | R1

LA04 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS 4.5.2.e.3 and Unit 2 4.5.2.f.3 each state "Verifying that upon receipt of an actual or simulated Recirculation Actuation Signal: each low-pressure safety injection pump stops, each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes." ITS does not contain details for automatic valve positioning in response to an actuation signal. ITS SR 3.5.2.6 and ITS SR 3.5.2.8 demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SIAS and on an RAS, and that the LPSI pumps stop on receipt of an actual or simulated RAS. This changes the CTS by deleting the details that state, in part, that upon receipt of an actual or simulated Recirculation Actuation Signal, each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes.

The removal of these details for performing surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the Surveillance Requirements to demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SIAS and on an RAS. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 - Relaxation of Required Action)* CTS 3.5.2 Action a.2 states that with one ECCS subsystem inoperable for reasons other than condition a.1 (i.e., one LPSI subsystem inoperable), restore the inoperable subsystem to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time. ITS 3.5.2 Action B states, with "one or more" trains inoperable for reasons other than Condition A, restore the train(s) to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time. Additionally, ITS 3.5.2 Action D states that with less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, enter LCO 3.0.3 immediately. CTS does not provide an action for the condition with less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS subsystem.

This changes the CTS by allowing combinations of equipment in each train to be credited with meeting the ECCS safety function provided 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. For example, under the CTS an inoperable HPSI pump in one train and an inoperable LPSI pump in the other train would require an immediate entry into CTS 3.0.3. Under the ITS, the same condition would allow 72 hours before requiring a shutdown because the remaining OPERABLE HPSI pump and LPSI pump are capable of producing flow equivalent to a single OPERABLE ECCS train.

The purpose of CTS 3.5.2 Action a.2 is to limit the period of time the plant can operate without redundant ECCS trains. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to degraded conditions in order to minimize risk associated with continued plant operation while providing time to repair inoperable equipment. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABILITY status of the remaining ECCS equipment, a reasonable time for repairs or replacement of required equipment, and the low probability of a design basis accident occurring during the time period. ITS 3.5.2 Action B continues to ensure that the capability of the equivalent of an ECCS train is available and limits the time in this condition to 72 hours. The ECCS in this condition can still perform its safety function, assuming no single failure occurs. ITS 3.5.2 Action D continues to ensure that there are necessary actions to place the plant in a safe condition outside of the Mode of Applicability since the ECCS safety function for 100% ECCS flow equivalent to a single ECCS train is no longer available. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 9 – Deletion of Reporting Requirements)* Unit 1 CTS 3.5.2 Action b. states "In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date." Unit 2 CTS 3.5.2 Action b. states "In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be

prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70." ITS 3.5.2 does not contain this CTS 90-day Special Report requirement in the event the ECCS is actuated and injects water into the RCS. ITS 3.5.2 retains the ACTIONS for inoperable ECCS trains. This changes the CTS by deleting this 90-day Special Report.

The removal of these details for making reports from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the LCO for two ECCS trains to be OPERABLE and ACTIONS when one or more ECCS trains are inoperable. Deletion of the ECCS reporting requirement is acceptable since 10 CFR 50.72(b)(2)(iv) requires a 4 hour notification in the event of an unplanned ECCS actuation and injection into the RCS. Additionally, 10 CFR 50.73(2)(iv)(A) also requires submission of a licensee event report for any unplanned event or condition that resulted in manual or automatic actuation of ECCS. PSL Unit 1 and Unit 2 operating licenses require compliance with 10 CFR 50.72 and 50.73. Therefore, the CTS requirement is unnecessary. Requirements continue to be provided in ITS to mitigate a DBA while in MODES 1, 2 and 3 with pressurizer pressure ≥ 1750 psia. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L03 *(Category 8 – Deletion of Surveillance Requirement Shutdown Performance Requirements)* Unit 1 CTS 4.5.2.e and Unit 2 CTS 4.5.2.f contain the requirement to perform the test "during shutdown." ITS SR 3.5.2.6, SR 3.5.2.7, and SR 3.5.2.8 do not specify conduct of the test to occur during shutdown. This changes the CTS by deleting the requirement to perform these Surveillances during shutdown. | R1

The purpose of Unit 1 CTS 4.5.2.e and Unit 2 CTS 4.5.2.f is to demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SIAS and on an RAS, that each ECCS pump starts on receipt of an actual or simulated SIAS, and that the LPSI pumps stop and each automatic containment sump, RWT outlet, and SI recirc valve actuates to the required position on receipt of an actual or simulated RAS.

The proposed Surveillance does not include the restriction that it be performed during shutdown. The control of unit conditions appropriate to perform the test is an issue for procedures and scheduling and has been determined by the NRC Staff to be unnecessary as a Technical Specification restriction. As indicated in Generic Letter 91-04, removal of this specific restriction is consistent with the vast majority of other Technical Specification Surveillances that do not dictate unit conditions for the Surveillance. This change is designated as less restrictive because the Surveillance may be performed during plant conditions other than shutdown.

L04 *(Category 5 – Deletion of Surveillance Requirement)* Unit 1 CTS 4.5.2.e.1 and Unit 2 CTS 4.5.2.f.1 require verification that each automatic ECCS valve in the

flow path actuates to its correct position. ITS SR 3.5.2.6 requires verification that each automatic ECCS valve in the flow path "that is not locked, sealed, or otherwise secured in position" actuates to the correct position. This changes the CTS by deleting the Surveillance Requirement for those valves that are locked, sealed, or otherwise secured in position.

The purpose of Unit 1 CTS 4.5.2.e.1 and Unit 2 CTS 4.5.2.f.1 is to provide assurance that if an event occurred requiring ECCS valves to be in their correct position, then those requiring automatic actuation would actuate to their correct position. This change is acceptable because the deleted Surveillance is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to provide confidence that the equipment can perform its assumed safety function. Those automatic valves that are locked, sealed, or otherwise secured in position are not required to actuate on a safety injection actuation signal or a recirculation actuation signal in order to perform their safety function because they are already in the required position. Testing such valves would not provide any additional assurance of OPERABILITY. Valves that are required to actuate will continue to be tested. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L05 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.5.2.e.1 and e.2 and Unit 2 4.5.2.f.1 and f.2 state, in part, that automatic valve actuations (e.1, f.1) and automatic pump actuations (e.2, f.2) occur upon receipt of a Safety Injection Actuation Signal. ITS SR 3.5.2.6 and SR 3.5.2.7 allow the use of a simulated or actual signal when performing these tests. This changes the CTS by allowing the use of unplanned actuations to perform the Surveillance based on the collection of sufficient information to satisfy the surveillance test requirements.

The purpose of CTS is to demonstrate that each automatic ECCS valve actuates to the required position on an SIAS and on an RAS, that each ECCS pump starts on receipt of an SIAS, and that the LPSI pumps stop on receipt of an RAS. This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal. Therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change is designated as less restrictive because it allows an actual signal to be credited for Surveillance where only a simulated signal was previously allowed in the CTS.

L06 **Unit 2 Only:** *(Category 7 - Relaxation of Surveillance Frequency)* CTS 4.5.2.h.1 states, in part, that verification of the correct position of each electrical and/or mechanical position stop for ECCS throttle valves is required “during valve stroking operation or following maintenance on the valve and prior to declaring the valve OPERABLE when the ECCS subsystems are required to be OPERABLE.” ISTS SR 3.5.2.1 states “Verify, for each ECCS throttle valve listed below, each position stop is in the correct position.” The ITS Surveillance does not require the valve position stop verification during valve stroking operation or following maintenance on the valve. This changes the CTS by deleting the Surveillance frequency requirement “during valve stroking operation or following

maintenance on the valve and prior to declaring the valve OPERABLE when the ECCS subsystems are required to be OPERABLE."

The purpose of CTS 4.5.h.1 is verification of the correct position of each electrical and/or mechanical position stop for ECCS throttle valves. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L07 **Unit 2 Only:** *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.5.2.a states in part, that each ECCS subsystem shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that the following valves are in the indicated positions with power to the valve operators removed. These valves include Safety Injection Tank (SIT) Vent Valves V3733, V3734, V3735, V3736, V3737, V3738, V3739, V3740. CTS 3.5.2 (ITS LCO 3.5.2) requires two ECCS subsystems to be OPERABLE with each subsystem comprised of one high-pressure safety injection (HPSI) pump, one low-pressure safety injection pump, flow path taking suction from the refueling water tank, and one charging pump. CTS 3.5.2 does not include the SITs as components of an ECCS subsystem. The SIT ECCS function is addressed in CTS 3.5.1 (ITS LCO 3.5.1). Neither ITS 3.5.1 nor 3.5.2 contains a Surveillance to verify the SIT vent valves are locked closed with power to the valve operators removed. This changes the CTS by deleting the Surveillance of verifying SIT vent valves are locked closed with power to the valve operators removed.

The purpose of CTS 4.5.2.a is to verify SIT vent valves are locked in the correct position with power to the valve operators removed to ensure a valve mispositioning event cannot occur, rendering the ECCS inoperable. The basis for this Surveillance, as described in the Bases of ISTS SR 3.5.2.1, is to ensure an inadvertent valve misalignment or active failure, similar to those described in IE Information Notice No. 87-01, does not result in disabling the function of both ECCS trains (subsystems) and invalidate the accident analysis. The SIT vent valves are not required to meet the LCO of CTS 3.5.2 and do not support the OPERABILITY of the ECCS subsystems. The SIT vent valves support OPERABILITY of the SITs by maintaining the integrity of the SIT pressure boundary. This change is acceptable because during a plant startup, the SIT vent valves must be closed to pressurize the SITs and meet the nitrogen cover pressure requirements of CTS 3.5.1.d and 4.5.1.1.a.1. This ensures the SITs are pressurized and available to inject its contents in the event of a LOCA. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Additionally, the SIT vent valves are designed to fail close on loss of power and a vent valve misposition will only affect one SIT. Therefore, an inadvertent valve misalignment or active failure, similar to those described in IE Information Notice No. 87-01, will not result in disabling the SIT ECCS function. Any SIT vent valve misposition will also result in a decrease of nitrogen cover pressure below the CTS 3.5.1.d (ITS SR 3.5.1.3) requirement requiring the associated SIT to be declared inoperable and appropriate actions

performed. Therefore, the deleted Surveillance Requirement is not necessary to support OPERABILITY of the ECCS subsystems and is not necessary to ensure that the SITs can perform their required ECCS function~~s~~.

This change is designated as less restrictive because a Surveillance~~s~~ which is required in the CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

3.5.2

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

3.5.2 Applicability

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure ≥ ~~[1700]~~ 1750 psia. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~----REVIEWER'S NOTE----~~ ~~The adoption of this Condition is contingent upon implementation of a program to perform a contemporaneous assessment of the overall impact on safety of proposed plant configurations prior to performing and during performance of maintenance activities that remove equipment from service.~~ ~~----------~~ 3		
ACTION 3.5.2.a.1 A. One LPSI subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~ 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION 3.5.2.a.2 DOC L01 B. One or more trains inoperable for reasons other than Condition A.	B.1 Restore train(s) to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
ACTION 3.5.2.a.1, a.2 C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Reduce pressurizer pressure to < ~~[1700]~~ 1750 psia.	6 hours 12 hours
DOC L01 D. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.5.2.a SR 3.5.2.1	~~[~~Verify the following valves are in the listed position with power to the valve operator removed ~~[and key locked in position]~~. Valve Number / Position / Function ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ V3659 Open Mini-Flow Isolation V3660 Open Mini-Flow Isolation	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.5.2.b.1 Note	SR 3.5.2.2	----------NOTE---------- Not required to be met for system vent flow paths opened under administrative control. ----------		
4.5.2.b.1		Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.5.2.b.2	SR 3.5.2.3	Verify ECCS locations susceptible to gas accumulation are sufficiently filled with water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.5.2.f	SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM	
DOC M01	SR 3.5.2.5	~~[~~Verify each charging pump develops a flow of ≥ ~~[36]~~ gpm ~~at a discharge pressure of ≥ [2200] psig~~. 40	In accordance with the INSERVICE TESTING PROGRAM ~~]~~	2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.5.2.e.1	SR 3.5.2.6	Verify each ECCS automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.5.2.e.2	SR 3.5.2.7	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.5.2.e.3	SR 3.5.2.8	Verify each LPSI pump stops on an actual or simulated actuate on signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.5.2.d.2	SR 3.5.2.9	~~[Verify, for each ECCS throttle valve listed below, each position stop is in the correct position.~~ ~~Valve Number~~ Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~	4 4
M02	SR 3.5.2.10	Verify each boric acid makeup tank water volume is ≥ 4000 gallons.	In accordance with the Surveillance Frequency Control Program	5
M02	SR 3.5.2.11	Verify each boric acid makeup tank boron concentration is ≥ 1900 ppm.	In accordance with the Surveillance Frequency Control Program	5

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

3.5.2

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

3.5.2 Applicability

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure ≥ ~~[1700]~~ 1750 psia. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~----REVIEWER'S NOTE----~~ ~~The adoption of this Condition is contingent upon implementation of a program to perform a contemporaneous assessment of the overall impact on safety of proposed plant configurations prior to performing and during performance of maintenance activities that remove equipment from service.~~ ~~----------~~ 3		
ACTION 3.5.2.a.1 A. One LPSI subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~ 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION 3.5.2.a.2 DOC L01 B. One or more trains inoperable for reasons other than Condition A.	B.1 Restore train(s) to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
ACTION 3.5.2.a.1, a.2 C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Reduce pressurizer pressure to < ~~[1700]~~ 1750 psia.	6 hours 12 hours
DOC L01 D. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.5.2.d.2 SR 3.5.2.1 ~~[Verify the following valves are in the listed position with power to the valve operator removed [and key locked in position]~~. ~~Valve Number~~ ~~Position~~ ~~Function~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ ~~[ ]~~ Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ ~~]~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.5.2.b Note	SR 3.5.2.2	----------NOTE---------- Not required to be met for system vent flow paths opened under administrative control. ----------		
4.5.2.b		Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	[31 days OR In accordance with the Surveillance Frequency Control Program]	2
4.5.2.c	SR 3.5.2.3	Verify ECCS locations susceptible to gas accumulation are sufficiently filled with water.	[31 days OR In accordance with the Surveillance Frequency Control Program]	2
4.5.2.g	SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM	
DOC M01	SR 3.5.2.5	[Verify each charging pump develops a flow of ≥ [36] gpm at a discharge pressure of ≥ [2200] psig. 40	In accordance with the INSERVICE TESTING PROGRAM]	2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.5.2.f.1	SR 3.5.2.6	Verify each ECCS automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.5.2.f.2	SR 3.5.2.7	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.5.2.f.3	SR 3.5.2.8	Verify each LPSI pump stops on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.5.2.h DOC L06	SR 3.5.2.9	~~[~~ Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. ~~Valve Number~~ ~~[ ]~~ ~~[ ]~~ HPSI System Valve Number: HCV 3616/3617, HCV 3626/3627, HCV 3636/3637, HCV 3646/3647, V3523/V3540 LPSI System Valve Number: HCV 3615, HCV 3625, HCV 3635, HCV 3645	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~	2 2
M02	SR 3.5.2.10	Verify each boric acid makeup tank water volume is ≥ 4000 gallons.	In accordance with the Surveillance Frequency Control Program	5
M02	SR 3.5.2.11	Verify each boric acid makeup tank boron concentration is ≥ 1900 ppm.	In accordance with the Surveillance Frequency Control Program	5

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.2, ECCS - OPERATING

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal. The adoption of this Condition is contingent upon implementation of a program to perform a contemporaneous assessment of the overall impact on safety of proposed plant configurations prior to performing and during performance of maintenance activities that remove equipment from service. St. Lucie Plant Unit 1 and Unit 2 Current Technical Specifications (CTS) were revised in Amendments 247 and 199, for Unit 1 and Unit 2 respectively, are revised to adopt a Risk Informed Completion Time (RICT) Program on July 2, 2019 (ADAMS Accession No. ML19113A099).

4. ISTS does not provide a Surveillance Requirement for visual inspection of the ECCS containment sump suction inlets for debris and the suction inlet trash racks and screens for evidence of structural distress or abnormal corrosion. Unit 1 CTS 4.5.2.d.2 and Unit 2 CTS 4.5.2.e.2 Surveillances requiring a containment sump visual inspection of ECCS pump suction inlets, are similar to the visual inspection required by ISTS SR 3.6.13.1. However, FPL is not adopting the ISTS 3.6.13 requirements but rather is retaining the current Surveillance. The ISTS is changed by adding ITS SR 3.2.5.9 for Unit 1 and SR 3.5.2.1 for Unit 2. ISTS SR 3.5.2.9 does not apply to PSL Unit 1 since mechanical stops are not required in the Unit 1 licensing basis. The Unit 1 ECCS design includes flow limiting orifices. Therefore, the ISTS SR 3.5.2.9 text is replaced with the proposed Surveillance Requirement in Unit 1 ITS. ISTS SR 3.5.2.1 does not apply for PSL Unit 2 since there is no single valve that could be mispositioned that would result in a loss of both ECCS trains as described in the ISTS Bases of SR 3.5.2.1. Therefore, the ISTS SR 3.5.2.1 text is replaced with the proposed Surveillance Requirement in the Unit 2 ITS.

5. ISTS 3.5.2 does not contain surveillances for the boric acid makeup tank (BAMT) volume and boron concentration for the charging pump ECCS function. The ISTS 3.5.2 assumes the refueling water tank (RWT) as the ECCS water source for low pressure safety injection, high pressure safety injection, and charging ECCS subsystems. The PSL design connects the ECCS charging subsystem to the BAMTs. Though the charging subsystem can be aligned to the RWT, and the charging pump suction automatically aligns to the RWT on Volume Control Tank low level, the RWT to charging pump suction valve gets a close signal on a safety injection actuation signal. This change adds SR 3.5.2.10 to verify that each BAMT volume is greater than or equal to the specified minimum required volume in accordance with the Surveillance Frequency Control Program and adds SR 3.5.2.11 to verify that each BAMT minimum concentration is greater than or equal to the specified minimum required boron concentration in accordance with the Surveillance Frequency Control Program. These new Surveillances are necessary to periodically verify that a sufficient usable BAMT water volume and boron concentration are available to the charging pump for injection during a small break LOCA. DOC M02

provides additional discussion of change for adding SR 3.5.2.10 and its Frequency and adding SR 3.5.2.11 and its Frequency.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA),

b. Control Element Assembly (CEA) ejection accident,

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater, and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are two phases of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the Reactor Coolant System (RCS) via the cold legs. ~~After the blowdown stage of the LOCA stabilizes, injection flow is split equally between the hot and cold legs.~~ After the refueling water tank (RWT) has been depleted, the ECCS recirculation phase is entered as the ECCS suction is automatically transferred to the containment sump.

1

Two redundant, 100% capacity trains are provided. In MODES 1, 2, and 3, with pressurizer pressure ≥ ~~1700~~ 1750 psia, each train consists of high pressure safety injection (HPSI), low pressure safety injection (LPSI), and charging subsystems. In MODES 1, 2, and 3, with pressurizer pressure ≥ ~~1700~~ 1750 psia, both trains must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided in the event of a single active failure.

1

1

A suction header supplies water from the RWT or the containment sump to the ECCS pumps. Separate piping supplies each train. The discharge headers from each HPSI pump divide into four supply lines. Both HPSI trains feed into each of the four injection lines. The discharge header from each LPSI pump ~~divides into two supply lines~~ combine to a common header, then divide into four injection lines, each feeding ~~the~~ one ~~injection line to two~~ RCS cold leg~~s~~. Control valves or orifices are set to balance the flow to the RCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

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1

BASES

BACKGROUND (continued)

For LOCAs that are too small to initially depressurize the RCS below the shutoff head of the HPSI pumps, the charging pumps supply water to maintain inventory until the RCS pressure decreases below the HPSI pump shutoff head. During this period, the steam generators (SGs) must provide the core cooling function. The charging pumps take suction from the ~~RWT~~ on a safety injection actuation signal (SIAS) and discharge directly to the RCS through a common header. ~~The normal supply source for the charging pumps is isolated on an SIAS to prevent noncondensible gas (e.g., air, nitrogen, or hydrogen) from being entrained in the charging pumps.~~

boric acid makeup tanks (BAMTs) via gravity feed valves or boric acid makeup pump discharge valves

1 4

During low temperature conditions in the RCS, limitations are placed on the maximum number of HPSI pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

During a large break LOCA, RCS pressure will decrease to < 200 psia in < 20 seconds. The safety injection (SI) systems are actuated upon receipt of an SIAS. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the diesel generators (DGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

1

E

emergency

1

and

The active ECCS components, along with the passive safety injection tanks (SITs)~~,~~ the RWT~~, and the containment sump, are~~ covered in LCO 3.5.1, "Safety Injection Tanks (SITs)," LCO 3.5.4, "Refueling Water Tank (RWT)," ~~and LCO 3.8.13, "Containment Sump,"~~ provide the cooling water necessary to meet GDC 35 (Ref. 1).

1

3

and

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria, established by 10 CFR 50.46 (Ref. 2) for ECCSs, will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F,

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

d. Core is maintained in a coolable geometry, and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following a steam line break (SLB) and ensures that containment temperature limits are met.

Both HPSI and LPSI subsystems are assumed to be OPERABLE in the large break LOCA analysis at full power (Ref. 3). This analysis establishes a minimum required runout flow for the HPSI and LPSI pumps, as well as the maximum required response time for their actuation. The HPSI pumps and charging pumps are credited in the small break LOCA analysis. This analysis establishes the flow and discharge head requirements at the design point for the HPSI pump. The SGTR and SLB analyses also credit the HPSI pumps, but are not limiting in their design.

The large break LOCA event with a loss of offsite power and a single failure (disabling one ECCS train) establishes the OPERABILITY requirements for the ECCS. During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control element assembly (CEA) insertion during small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

On smaller breaks, RCS pressure will stabilize at a value dependent upon break size, heat load, and injection flow. The smaller the break, the higher this equilibrium pressure. In ~~all~~ LOCA analyses, injection flow is not credited until RCS pressure drops below the shutoff head of the HPSI pumps. 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO ensures that an ECCS train will deliver sufficient water to match decay heat boiloff rates soon enough to minimize core uncovery for a large LOCA. It also ensures that the HPSI pump will deliver sufficient water during a small break LOCA and provide sufficient boron to maintain the core subcritical following an SLB. For smaller LOCAs, the charging pumps deliver sufficient fluid to maintain RCS inventory until the RCS can be depressurized below the HPSI pumps' shutoff head. During this period of a small break LOCA, the SGs continue to serve as the heat sink providing core cooling.

ECCS - Operating satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODES 1, 2, and 3, with pressurizer pressure ≥ ~~1700~~ 1750 psia, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming there is a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents. 1

In MODES 1 and 2, and in MODE 3 with pressurizer pressure ≥ ~~1700~~ 1750 psia, an ECCS train consists of an HPSI subsystem, an LPSI subsystem, and a charging pump. 1

Each ~~train~~ HPSI and LPSI subsystem includes the piping, instruments, and controls to ensure the availability of an OPERABLE flow path capable of taking suction from the RWT on an SIAS and automatically transferring suction to the containment sump upon a recirculation actuation signal (RAS). Management of gas voids is important to ECCS OPERABILITY. 1

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the RWT to the RCS, via the HPSI and LPSI pumps and their respective supply headers, to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply part of its flow to the RCS hot legs via the shutdown cooling (SDC) suction nozzles. ~~The~~ charging pump flow path take~~s~~ s suction from the ~~RWT~~ BAMT(s) and supplies the RCS via the normal charging lines. 1

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

One charging pump flow path taking suction from either a single BAMT via the associated boric acid makeup pump, or both BAMTs via both boric acid makeup pumps. A second charging pump flow path taking suction from either a single BAMT via gravity feed valves, or both BAMTs via gravity feed valves. 4

BASES

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure ≥ ~~1700~~ 1750 psia, the ECCS OPERABILITY requirements for the limiting Design Basis Accident (DBA) large break LOCA are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPSI pump performance is based on the small break LOCA, which establishes the pump performance curve and has less dependence on power. The charging pump performance requirements are based on a small break LOCA. The requirements of MODES 2 and 3, with RCS pressure ≥ ~~1700~~ 1750 psia, are bounded by the MODE 1 analysis. 1

The ECCS functional requirements of MODE 3, with RCS pressure < ~~1700~~ 1750 psia, and MODE 4 are described in LCO 3.5.3, "ECCS - Shutdown." 1

In MODES 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one LPSI subsystem inoperable, action must be taken to restore OPERABLE status within 7 days ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. In this condition, the remaining OPERABLE ECCS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining LPSI subsystem could result in loss of ECCS function. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable LPSI subsystem. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 6. Reference 6 concluded that extending the Completion Time to 7 days for an inoperable LPSI train provides plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the LPSI train unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations. 2

BASES

ACTIONS (continued)

B.1

If one or more trains are inoperable except for reasons other than Condition A (one LPSI subsystem inoperable) and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available, the inoperable components must be returned to OPERABLE status within 72 hours [or in accordance with the Risk Informed Completion Time Program]. The 72 hour Completion Time is based on an NRC study (Ref. 4) using a reliability evaluation and is a reasonable amount of time to effect many repairs. 2

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS. The individual components are inoperable if they are not capable of performing their design function, or if supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in plant operations when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an emergency DG can disable one ECCS train until power is restored. A reliability analysis (Ref. 4) has shown that the impact with one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 5 describes situations in which one component, such as a shutdown cooling total flow control valve, can disable both ECCS trains. With one or more components inoperable, such that 100% of the equivalent flow to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

BASES

ACTIONS (continued)

C.1 and C.2

If the inoperable train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to <~~1700~~ psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems.

1750

1

D.1

D is applicable with less than 100% of the ECCS flow equivalent to a single OPERABLE train available.

Condition ~~B is applicable with one or more trains inoperable~~. The allowed Completion Time is based on the assumption that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the facility is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

of Condition B

5

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removing power ~~or by key locking the control in the correct position~~ ensures that the valves cannot be inadvertently misaligned or change position as the result of an active failure. These valves are of the type described in Reference 5, which can disable the function of both ECCS trains and invalidate the accident analysis. ~~[A 12 hour Frequency is considered reasonable in view of other administrative controls ensuring that a mispositioned valve is an unlikely possibility.~~

The two series minimum flow isolation valves are common to the HPSI, LPSI, and containment spray pumps. Therefore, it is critical that these valves must be maintained open during the initial phase of a LOCA to prevent ECCS pump damage as a result of the pumps operating at shutoff head.

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2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve automatically repositions within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position.

[The 31 day Frequency is appropriate because the valves are operated under procedural control and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

2

2

The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent flow path who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the system vent flow path if directed.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3

ECCS piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the ECCS and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

Selection of ECCS locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The ECCS is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the ECCS is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits. | R1

ECCS locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and

BASES

SURVEILLANCE REQUIREMENTS (continued)

determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the adequacy of the procedural controls governing system operation.~~
~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the unit safety analysis. SRs are specified in the INSERVICE TESTING PROGRAM of the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5

~~Discharge head at~~ design flow is a normal test of charging pump performance required by the ASME Code and the INSERVICE TESTING PROGRAM. A quarterly Frequency for such tests is a Code requirement. Such inservice ~~inspections~~ tests detect component degradation and incipient failures.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.6, SR 3.5.2.7, and SR 3.5.2.8

These SRs demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SIAS and on an RAS, that each ECCS pump starts on receipt of an actual or simulated SIAS, and that the LPSI pumps stop on receipt of an actual or simulated RAS. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. [The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

and each automatic containment sump, RWT outlet, and SI recirc valve to the RWT actuates to the required position

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OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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The actuation logic is tested as part of the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the INSERVICE TESTING PROGRAM.

SR 3.5.2.9

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Realignment of valves in the flow path on an SIAS is necessary for proper ECCS performance. The safety injection valves have stops to position them properly so that flow is restricted to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. This SR is not required for units with flow limiting orifices. [The 18 month Frequency is based on the same factors as those stated above for SR 3.5.2.6, SR 3.5.2.7, and SR 3.5.2.8.

7

Periodic inspections of ECCS train containment sump suction inlets ensure they are unrestricted and stay in proper operating condition.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.

2. 10 CFR 50.46.

U

3. FSAR, Chapter ~~[~~6~~]~~. 2

4. NRC Memorandum to V. Stello, Jr., from R. L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

5. IE Information Notice No. 87-01, January 6, 1987.

6. CE NPSD-995, "Low Pressure Safety Injection System AOT Extension," May 1995.

SR 3.5.2.10

This Surveillance verifies that the water volume in the BAMT is above the minimum usable volume plus additional margin to ensure that a sufficient water supply is available to the charging pump for injection during a small break LOCA. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 8

SR 3.5.2.11

This Surveillance verifies that the boron concentration in the BAMT is above the minimum BAMT boron concentration to ensure that a sufficient borated water supply is available to the charging pump for injection during a small break LOCA. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 8

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA),

b. Control Element Assembly (CEA) ejection accident,

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater, and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are two phases of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the Reactor Coolant System (RCS) via the cold legs. ~~After the blowdown stage of the LOCA stabilizes, injection flow is split equally between the hot and cold legs.~~ After the refueling water tank (RWT) has been depleted, the ECCS recirculation phase is entered as the ECCS suction is automatically transferred to the containment sump. 1

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Two redundant, 100% capacity trains are provided. In MODES 1, 2, and 3, with pressurizer pressure ≥ ~~1700~~ psia, each train consists of high pressure safety injection (HPSI), low pressure safety injection (LPSI), and charging subsystems. In MODES 1, 2, and 3, with pressurizer pressure ≥ ~~1700~~ psia, both trains must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided in the event of a single active failure. 1 1

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A suction header supplies water from the RWT or the containment sump to the ECCS pumps. Separate piping supplies each train. The discharge headers from each HPSI pump divide into four supply lines. Both HPSI trains feed into each of the four injection lines. The discharge header from each LPSI pump divides into two supply lines, each feeding the injection line to two RCS cold legs. Control valves or orifices are set to balance the flow to the RCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

BASES

BACKGROUND (continued)

For LOCAs that are too small to initially depressurize the RCS below the shutoff head of the HPSI pumps, the charging pumps supply water to maintain inventory until the RCS pressure decreases below the HPSI pump shutoff head. During this period, the steam generators (SGs) must provide the core cooling function. The charging pumps take suction from the ~~RWT~~ on a safety injection actuation signal (SIAS) and discharge directly to the RCS through a common header. ~~The normal supply source for the charging pumps is isolated on an SIAS to prevent noncondensible gas (e.g., air, nitrogen, or hydrogen) from being entrained in the charging pumps.~~

boric acid makeup tanks (BAMTs) via gravity feed valves or boric acid makeup pump discharge valves

1 4

During low temperature conditions in the RCS, limitations are placed on the maximum number of HPSI pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

During a large break LOCA, RCS pressure will decrease to < 200 psia in < 20 seconds. The safety injection (SI) systems are actuated upon receipt of an SIAS. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the diesel generators (DGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

1

emergency

E

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and

The active ECCS components, along with the passive safety injection tanks (SITs)~~,~~ the RWT~~, and the containment sump, are~~ covered in LCO 3.5.1, "Safety Injection Tanks (SITs)," LCO 3.5.4, "Refueling Water Tank (RWT)," ~~and LCO 3.8.13, "Containment Sump,"~~ provide the cooling water necessary to meet GDC 35 (Ref. 1).

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3

and

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria, established by 10 CFR 50.46 (Ref. 2) for ECCSs, will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F,

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation,

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

d. Core is maintained in a coolable geometry, and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following a steam line break (SLB) and ensures that containment temperature limits are met.

Both HPSI and LPSI subsystems are assumed to be OPERABLE in the large break LOCA analysis at full power (Ref. 3). This analysis establishes a minimum required runout flow for the HPSI and LPSI pumps, as well as the maximum required response time for their actuation. The HPSI pumps and charging pumps are credited in the small break LOCA analysis. This analysis establishes the flow and discharge head requirements at the design point for the HPSI pump. The SGTR and SLB analyses also credit the HPSI pumps, but are not limiting in their design.

The large break LOCA event with a loss of offsite power and a single failure (disabling one ECCS train) establishes the OPERABILITY requirements for the ECCS. During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control element assembly (CEA) insertion during small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

On smaller breaks, RCS pressure will stabilize at a value dependent upon break size, heat load, and injection flow. The smaller the break, the higher this equilibrium pressure. In ~~all~~ LOCA analyses, injection flow is not credited until RCS pressure drops below the shutoff head of the HPSI pumps. 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO ensures that an ECCS train will deliver sufficient water to match decay heat boiloff rates soon enough to minimize core uncovery for a large LOCA. It also ensures that the HPSI pump will deliver sufficient water during a small break LOCA and provide sufficient boron to maintain the core subcritical following an SLB. For smaller LOCAs, the charging pumps deliver sufficient fluid to maintain RCS inventory until the RCS can be depressurized below the HPSI pumps' shutoff head. During this period of a small break LOCA, the SGs continue to serve as the heat sink providing core cooling.

ECCS - Operating satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

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In MODES 1, 2, and 3, with pressurizer pressure ≥ ~~1700~~ psia, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming there is a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

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In MODES 1 and 2, and in MODE 3 with pressurizer pressure ≥ ~~1700~~ psia, an ECCS train consists of an HPSI subsystem, an LPSI subsystem, and a charging pump.

1

HPSI and LPSI subsystem

Each ~~train~~ includes the piping, instruments, and controls to ensure the availability of an OPERABLE flow path capable of taking suction from the RWT on an SIAS and automatically transferring suction to the containment sump upon a recirculation actuation signal (RAS). Management of gas voids is important to ECCS OPERABILITY.

1

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the RWT to the RCS, via the HPSI and LPSI pumps and their respective supply headers, to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply part of its flow to the RCS hot legs via the shutdown cooling (SDC) suction nozzles. ~~The~~ charging pump flow path take~~s~~ suction from the ~~RWT~~ and supplies the RCS via the normal charging line~~s~~.

BAMT(s)

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

One charging pump flow path taking suction from either a single BAMT via the associated boric acid makeup pump, or both BAMTs via both boric acid makeup pumps. A second charging pump flow path taking suction from either a single BAMT via gravity feed valves, or both BAMTs via gravity feed valves.

4

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BASES

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure ≥ ~~1700~~ 1750 psia, the ECCS OPERABILITY requirements for the limiting Design Basis Accident (DBA) large break LOCA are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPSI pump performance is based on the small break LOCA, which establishes the pump performance curve and has less dependence on power. The charging pump performance requirements are based on a small break LOCA. The requirements of MODES 2 and 3, with RCS pressure ≥ ~~1700~~ 1750 psia, are bounded by the MODE 1 analysis. 1

The ECCS functional requirements of MODE 3, with RCS pressure < ~~1700~~ 1750 psia, and MODE 4 are described in LCO 3.5.3, "ECCS - Shutdown." 1

In MODES 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one LPSI subsystem inoperable, action must be taken to restore OPERABLE status within 7 days ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. In this condition, the remaining OPERABLE ECCS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining LPSI subsystem could result in loss of ECCS function. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable LPSI subsystem. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 6. Reference 6 concluded that extending the Completion Time to 7 days for an inoperable LPSI train provides plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the LPSI train unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations. 2

BASES

ACTIONS (continued)

B.1

If one or more trains are inoperable except for reasons other than Condition A (one LPSI subsystem inoperable) and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available, the inoperable components must be returned to OPERABLE status within 72 hours [or in accordance with the Risk Informed Completion Time Program]. The 72 hour Completion Time is based on an NRC study (Ref. 4) using a reliability evaluation and is a reasonable amount of time to effect many repairs. 2

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS. The individual components are inoperable if they are not capable of performing their design function, or if supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in plant operations when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an emergency DG can disable one ECCS train until power is restored. A reliability analysis (Ref. 4) has shown that the impact with one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 5 describes situations in which one component, such as a shutdown cooling total flow control valve, can disable both ECCS trains. With one or more components inoperable, such that 100% of the equivalent flow to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

BASES

ACTIONS (continued)

C.1 and C.2

If the inoperable train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < ~~1700~~ psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems.

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D.1

D is applicable with less than 100% of the ECCS flow equivalent to a single OPERABLE train available.

Condition ~~B is applicable with one or more trains inoperable~~. The allowed Completion Time is based on the assumption that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the facility is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

5

of Condition B

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

~~Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removing power or by key locking the control in the correct position ensures that the valves cannot be inadvertently misaligned or change position as the result of an active failure. These valves are of the type described in Reference 5, which can disable the function of both ECCS trains and invalidate the accident analysis. [A 12 hour Frequency is considered reasonable in view of other administrative controls ensuring that a mispositioned valve is an unlikely possibility.~~

7

~~OR~~ Periodic inspections of ECCS train containment sump suction inlets ensure they are unrestricted and stay in proper operating condition.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve automatically repositions within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position.

[The 31 day Frequency is appropriate because the valves are operated under procedural control and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent flow path who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the system vent flow path if directed.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3

ECCS piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the ECCS and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

Selection of ECCS locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The ECCS is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met, and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the ECCS is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits. | R1

ECCS locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and

BASES

SURVEILLANCE REQUIREMENTS (continued)

determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the adequacy of the procedural controls governing system operation.~~
~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the unit safety analysis. SRs are specified in the INSERVICE TESTING PROGRAM of the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5

~~Discharge head at~~ design flow is a normal test of charging pump performance required by the ASME Code and the INSERVICE TESTING PROGRAM. A quarterly Frequency for such tests is a Code requirement. Such inservice ~~inspections~~ tests detect component degradation and incipient failures.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.6, SR 3.5.2.7, and SR 3.5.2.8

These SRs demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SIAS and on an RAS, that each ECCS pump starts on receipt of an actual or simulated SIAS, and that the LPSI pumps stop on receipt of an actual or simulated RAS. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. [The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

and each automatic containment sump, RWT outlet, and SI recirc valve to the RWT actuates to the required position

6

2

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

]

2

The actuation logic is tested as part of the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the INSERVICE TESTING PROGRAM.

SR 3.5.2.9

Realignment of valves in the flow path on an SIAS is necessary for proper ECCS performance. The safety injection valves have stops to position them properly so that flow is restricted to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. This SR is not required for units with flow limiting orifices. [The 18 month Frequency is based on the same factors as those stated above for SR 3.5.2.6, SR 3.5.2.7, and SR 3.5.2.8.

The HPSI header injection valves are throttled to prevent HPSI pump runout.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.

2. 10 CFR 50.46.

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3. FSAR, Chapter ~~[~~6~~]~~. 2

4. NRC Memorandum to V. Stello, Jr., from R. L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

5. IE Information Notice No. 87-01, January 6, 1987.

6. CE NPSD-995, "Low Pressure Safety Injection System AOT Extension," May 1995.

SR 3.5.2.10

This Surveillance verifies that the water volume in the BAMT is above the minimum usable volume plus additional margin to ensure that a sufficient water supply is available to the charging pump for injection during a small break LOCA. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 8

SR 3.5.2.11

This Surveillance verifies that the boron concentration in the BAMT is above the minimum BAMT boron concentration to ensure that a sufficient borated water supply is available to the charging pump for injection during a small break LOCA. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 8

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.2, BASES, ECCS - OPERATING

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ISTS 3.6.13, "Containment Sump" is not being adopted and St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) does not have a separate containment sump Specification. The Containment Sump function ensures debris is not transported to the ECCS and containment spray pump suction inlets during a LOCA, which could inhibit operation of these mitigation systems. Requirements associated with this function are provided in ITS 3.5.2, "ECCS – Operating", and ITS 3.5.3, "ECCS – Shutdown" for PSL Unit 1 and Unit 2. Therefore, ISTS 3.6.13 is not included in the PSL Unit 1 and Unit 2 ITS. The reference to a containment sump Specification is deleted.

4. ISTS Background and LCO descriptions include charging pump details. These details are enhanced to include charging pump flow paths that maintain the required independence of ECCS trains. These details for the charging pump flow paths are relocated from the CTS and are added to the ITS Bases.

5. Editorial change made to the Bases of Condition D to reflect the wording of Condition D in ISTS 3.5.2 (ITS 3.5.2) and further clarify the distinction between Conditions B and D.

6. ISTS SR 3.5.5.6, SR 3.5.2.7, and SR 3.5.2.8 description contains that each automatic ECCS valve actuates and each ECCS pump starts on an actual or simulated SIAS and on an RAS, and that the LPSI pump stops on receipt of an actual or simulated RAS. Unit 1 CTS 4.5.2.e.3 and Unit 2 4.5.2.f.3 each provide details that upon receipt of an actual or simulated RAS, that in addition to each LPSI pump stops, each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes. These details for automatic valves design actuation in response to an RAS are relocated from the CTS and are added to the ISTS Bases to recognize these design features.

7. The CTS Surveillance requiring a containment sump visual inspection of ECCS pump suction inlets is added to ITS SR 3.2.5.9 for Unit 1 and SR 3.5.2.1 for Unit 2. The ISTS Bases are revised to reflect the change to the ISTS SRs. The Bases discussion added is similar to the discussion provided for the Bases of ISTS SR 3.5.2.10 in Revision 4 of NUREG-1432.

8. ISTS 3.5.2 does not contain surveillances for the boric acid makeup tank (BAMT) volume and boron concentration for the charging pump ECCS function. This ISTS Bases change reflects the change to the Specification, which adds SR 3.5.2.10 to verify that each BAMT volume is greater than or equal to the specified minimum volume in accordance with the Surveillance Frequency Control Program and SR 3.5.2.11 to verify that each BAMT minimum boron concentration is greater than or

equal to the specified minimum required boron concentration in accordance with the Surveillance Frequency Control Program. The added Bases discussion is similar to the Bases discussion provided for the refueling water tank (RWT) volume verification of ISTS SR 3.5.4.2 and the RWT boron concentration verification of ISTS SR 3.5.4.3.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.5.2, ECCS - OPERATING

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

3.5.3 ECCS - Shutdown

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

In MODE 4, only the safety injection actuation and recirculation actuation manual channels and associated actuation logic are required. — L01

high pressure safety injection (HPSI) train

LCO 3.5.3

~~3.5.3~~ ~~As a minimum,~~ one ~~ECCS subsystem comprised of the following~~ shall be OPERABLE:

~~a. In MODES 3* and 4#, one ECCS subsystem composed of one OPERABLE high pressure safety injection pump and one OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal.~~ — LA01

b. Prior to decreasing the reactor coolant system temperature below 270°F a maximum of only one high pressure safety injection pump shall be OPERABLE with its associated header stop valve open. — See ITS 3.4.12

c. Prior to decreasing the reactor coolant system temperature below 236°F all high pressure safety injection pumps shall be disabled and their associated header stop valves closed except as allowed by Specifications 3.1.2.1 and 3.1.2.3.

Applicability

APPLICABILITY: MODES 3* and 4~~.~~ with RCS cold leg temperature > 250°F — A03

MODES 5 and 6 when the Pressurizer manway cover is in place and the reactor vessel head is on. — See ITS 3.4.12

ACTION:

Required HPSI train inoperable — required HPSI train — within 1 hour — L02

ACTION A

~~a. With no ECCS subsystems OPERABLE in MODES 3* and 4#, immediately~~ restore ~~one ECCS subsystem~~ to OPERABLE status or be in ~~COLD SHUTDOWN~~ within ~~20~~ 24 hours. — L03

ACTION B

MODE 4 with RCS cold leg temperature ≤ 250°F — A03

b. With RCS temperature below 270°F and with more than the allowed high pressure safety injection pump OPERABLE or injection valves and header isolation valves open, immediately disable the high pressure safety injection pump(s) or close the header isolation valves. — See ITS 3.4.12

~~c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.~~ — L04

ACTIONS Note

~~d.~~ LCO 3.0.4.b is not applicable to ECCS High Pressure Safety Injection subsystem when entering MODE 4.

SURVEILLANCE REQUIREMENTS

For required, HPSI train, applicable portions of the following SRs are applicable: SR 3.5.2.1 SR 3.5.2.2 SR 3.5.2.4 SR 3.5.2.7 SR 3.5.2.3 SR 3.5.2.6 SR 3.5.2.9 — A02

SR 3.5.3.1

~~4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.~~

4.5.3.2 The high pressure safety injection pumps shall be verified inoperable and the associated header stop valves closed prior to decreasing below the above specified Reactor Coolant System temperature and once per month when the Reactor Coolant System is at refueling temperatures. — See ITS 3.4.12

Applicability * With pressurizer pressure < 1750 psia.

Applicability # REACTOR COOLANT SYSTEM cold leg temperature above 250°F.

A01

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - SHUTDOWN

In MODE 4, only the safety injection actuation and recirculation actuation manual channels and associated actuation logic are required. — L01

LIMITING CONDITION FOR OPERATION

high pressure safety injection (HPSI) train

LCO 3.5.3 ~~3.5.3~~ ~~As a minimum,~~ one ~~ECCS subsystem comprised of the following~~ shall be OPERABLE:

~~a. One OPERABLE high-pressure safety injection pump, and~~

~~b. An OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Sump Recirculation Actuation Signal.~~

LA01

APPLICABILITY: MODES 3* and 4#.

Footnote # shall remain applicable in MODES 5 and 6 when the Pressurizer manway cover is in place and the reactor vessel head is on. — See ITS 3.4.12

ACTION:

Required HPSI train inoperable — required HPSI train

ACTION A a. ~~With no ECCS subsystems OPERABLE,~~ restore ~~at least one ECCS subsystem~~ to OPERABLE status within 1 hour or be in ~~COLD SHUTDOWN~~ within ~~the next~~ ~~20~~ hours.

Mode 5

ACTION B 24 — L03

~~b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~ — L04

ACTIONS Note ~~c.~~ LCO 3.0.4.b is not applicable to ECCS High Pressure Safety Injection subsystem when entering MODE 4.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1 ~~4.5.3~~ ~~The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.~~

For required, HPSI train, applicable portions of the following SRs are applicable:
SR 3.5.2.1 SR 3.5.2.4 SR 3.5.2.7
SR 3.5.2.2 SR 3.5.2.6 SR 3.5.2.9
SR 3.5.2.3

A02

* With pressurizer pressure less than 1750 psia.

\# One HPSI shall be rendered inoperable prior to entering MODE 5. — See ITS 3.4.12

DISCUSSION OF CHANGES
ITS 3.5.3, ECCS - SHUTDOWN

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 4.5.3.1, and Unit 2 CTS 4.5.3, state that the ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2, but do not specify the applicable Surveillance Requirements. Similarly, ITS SR 3.5.3.1 specifies the applicable ITS 3.5.2 Surveillance Requirements.

The purpose of this SR is to ensure the required HPSI train is OPERABLE. Per the definition of OPERABILITY, the required HPSI train is OPERABLE or has OPERABILITY when it is capable of performing its specified safety function and when all necessary attendant equipment required for the HPSI train to perform its specified safety function are also capable of performing their related support function(s). Thus, only the SRs required to verify the HPSI train can perform its intended safety function are required. SR 3.5.2.5 and SR 3.5.2.8 verify charging pump and LPSI pump features, respectively, and are not part of the required HPSI train. Therefore, these SRs are not specified in the ITS.

This change is designated as administrative and is acceptable because it only clarifies the current requirement and does not result in a technical change to the CTS.

A03 **Unit 1 only**: CTS 3.5.3 Applicability requires, in part, at least one ECCS subsystem to be OPERABLE in MODE 4. CTS 3.5.3.a describes an ECCS subsystem in MODE 4 as consisting of one OPERABLE HPSI pump with the proper flowpath. The MODE 4 requirement in CTS 3.5.3.a is modified by # footnote, which states Reactor Coolant System cold leg temperature above 250°F. In MODE 4 with RCS cold leg temperature above 250°F (per # footnote), CTS 3.5.3 Action a requires the plant to be in Cold Shutdown (i.e., MODE 5) within the required completion time if the required ECCS subsystem cannot be restored immediately. ITS 3.5.3 requires one HPSI train to be OPERABLE in MODE 3 with pressurizer pressure < 1750 psia and in MODE 4 with RCS cold leg temperature > 250°F. When the required HPSI train (i.e., required ECCS subsystem) cannot be restored to OPERABLE status within the required Completion Time, ITS 3.5.3 Required Action B.1 requires the unit to be in MODE 4 with RCS cold leg temperature ≤ 250°F. This changes the Applicability from MODE 4 to MODE 4 with RCS cold leg temperature > 250°F and the action end state from MODE 4 to MODE 4 with RCS cold leg temperature ≤ 250°F.

The CTS 3.5.3 Applicability has a dual purpose: to provide applicability for the ECCS shutdown requirements and the low temperature overpressure protection (LTOP) requirements. ITS 3.5.3 provides ECCS shutdown requirements and ITS

3.4.12 provides LTOP requirements. The purpose of the ECCS shutdown requirements is to ensure adequate core cooling systems are available in the event of a loss of RCS inventory when the reactor is shutdown and the RCS is hot and pressurized. The purpose of CTS 3.5.3 Action a is to restore the required ECCS subsystem to OPERABLE status in a timely manner or place the plant in a condition in which the equipment is no longer required. CTS 3.5.3.a and Action a address the requirement for one ECCS subsystem when in MODE 3 with pressurizer pressure < 1750 psia and in MODE 4 with RCS cold leg temperature > 250°F. In MODE 4 below 250°F there is no CTS requirement or action related to ECCS. CTS 3.5.3.b and c and the associated action support the LTOP analysis and are not related to the ECCS function. Specifically, CTS 3.5.3.b limits HPSI to a maximum of only one with its associated header stop valve open when RCS temperature is ≤ 270°F and CTS 3.5.3.c requires all HPSI pumps to be disabled and their header stop valves closed (i.e., incapable of injecting into the RCS) when RCS temperature is ≤ 236°F in MODE 4. As a result, no ECCS is required to be OPERABLE in MODE 4 with RCS cold leg temperature ≤ 250°F. Therefore, the ITS ECCS Shutdown Applicability is modified from MODE 4 to MODE 4 with RCS cold leg temperature ≤ 250°F consistent with the CTS intent and the end state action is modified to require exiting the Applicability consistent with the requirements of CTS 3.0.2 (ITS LCO 3.0.2).

These changes are designated as administrative changes and are acceptable because they maintain the current requirement that one HPSI train be OPERABLE in MODE 4 with RCS cold leg temperature > 250°F and to place the unit in a condition in which the required HPSI train is no longer required consistent with the requirements of CTS 3.0.2 (ITS LCO 3.0.2) and, therefore, do not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.5.3.a and Unit 2 CTS 3.5.3.a and 3.5.3.b, state, in part, that as a minimum, one ECCS subsystem composed of one OPERABLE high pressure safety injection pump and one OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal. Additionally, Discussion of Change L01 changes the CTS 3.5.3 LCO to only require manual channel actuation in MODE 4. ITS 3.5.3 retains the requirement that one high pressure safety

injection (HPSI) train shall be OPERABLE. This changes the CTS by moving the details of what composes an OPERABLE ECCS subsystem, including flowpath and actuation functions, to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that one high pressure safety injection (HPSI) train shall be OPERABLE. Also this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specifications Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.5.3 states, in part, that one OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal is required to be OPERABLE in MODE 3 with pressurizer pressure < 1750 psia and MODE 4. This changes the CTS 3.5.3 LCO to only require manual channel actuation in MODE 4. CTS 3.3.2.1 and Table 3.3-3 only require automatic SIAS and RAS in MODES 1, 2, and 3, and manual channel actuation is required in MODE 4. Furthermore, Discussion of Change LA01 changes the CTS by moving the details of what composes an OPERABLE ECCS subsystem, including flowpath and actuation functions, to the Bases.

The purpose of CTS 3.5.3 is to state the limiting conditions for operation. This change is acceptable because the LCO requirements continue to ensure that the required HPSI train is maintained consistent with the safety analyses and licensing basis. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L02 **Unit 1 Only:** *(Category 3 – Relaxation of Completion Time)* CTS 3.5.3 Action a. states, in part, “immediately restore one ECCS subsystem to OPERABLE status or be in COLD SHUTDOWN within 20 hours.” With the required HPSI train inoperable, ITS 3.5.3 Required Action A.1 allows 1 hour to restore the required HPSI train to OPERABLE status. This changes the CTS by relaxing the Completion Time from “immediately” to 1 hour.

The purpose of CTS 3.5.3 Action a. is to take action with no HPSI train OPERABLE. With no HPSI train OPERABLE, the unit is not prepared to respond to a loss of coolant accident. The ITS 1 hour Completion Time to restore at least one HPSI train to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity or to initiate actions to place the unit in a condition where an ECCS train is not required. This change is acceptable

because the Completion Time is consistent with the Completion Time in the ISTS and continues to provide safe operation under the specific Condition, considering the low probability of a DBA occurring during the allowed Completion Time.

This change is designated as less restrictive because additional time is allowed to restore the required HPSI train to OPERABLE status than was allowed in the CTS.

L03 *(Category 3 – Relaxation of Completion Time)* Unit 1 CTS 3.5.3 Action a. states, in part, "immediately restore one ECCS subsystem to OPERABLE status or be in COLD SHUTDOWN within 20 hours." Unit 2 CTS 3.5.3 Action a. states, in part, "restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours." With the Required Action and associated Completion Time not met, that being the required HPSI train is not restored to OPERABLE status within the associated Completion Time, ITS 3.5.3 Required Action B.1 allows 24 hours to be in a condition where an ECCS train is not required. This changes the CTS by relaxing the Completion Time from "within the next 20 hours" to 24 hours.

The purpose of CTS 3.5.3 Action a. is to take action with no HPSI train OPERABLE. With no HPSI train OPERABLE, the unit is not prepared to respond to a loss of coolant accident. The ITS 24 hour Completion Time for ITS 3.5.3 Required Action B.1 is reasonable, based on operating experience, to reach the condition in an orderly manner and without challenging plant systems. This change is acceptable because the Completion Time is consistent with the time in the ISTS to reach the required plant condition to provide safe operation under the specific Condition, considering the low probability of a DBA occurring during the allowed Completion Time.

This change is designated as less restrictive because additional time is allowed to place the unit in a MODE or specified condition where an ECCS train is not required than was allowed in the CTS.

L04 *(Category 9 – Deletion of Reporting Requirements)* Unit 1 CTS 3.5.3 Action c. states "In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date." Unit 2 CTS 3.5.3 Action b. states "In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70." ITS 3.5.3 does not contain this CTS 90-day Special Report requirement in the event the ECCS is actuated and injects water into the RCS. ITS 3.5.3 retains the ACTIONS for the required HPSI train inoperable. This changes the CTS by deleting this 90-day Special Report.

The removal of these details for making reports from the Technical Specifications is acceptable because this type of information is not necessary to be included in

the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the LCO for one required HPSI train OPERABLE and the ACTIONS for the required HPSI train inoperable. Deletion of the ECCS reporting requirement is acceptable since 10 CFR 50.72(b)(2)(iv) requires a 4 hour notification in the event of an unplanned ECCS actuation and injection into the RCS. Additionally, 10 CFR 50.73(2)(iv)(A) also requires submission of a licensee event report for any unplanned event or condition that resulted in manual or automatic actuation of ECCS. PSL Unit 1 and Unit 2 operating licenses require compliance with 10 CFR 50.72 and 50.73. Therefore, the CTS requirement is unnecessary. Requirements continue to be provided in ITS to mitigate a DBA while Mode 3 with pressurizer pressure < 1750 psia and in Hot Standby (MODE 4) conditions. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3

LCO 3.5.3 One high pressure safety injection (HPSI) train shall be OPERABLE.

3.5.3 Applicability
MODE 3 Footnote*
MODE 4 Footnote#

APPLICABILITY: MODE 3 with pressurizer pressure < 1750 ~~[1700]~~ psia, (2)
MODE 4 with RCS cold leg temperature > 250°F (4)

ACTIONS

---NOTE---

ACTION 3.5.3.d

LCO 3.0.4.b is not applicable to ECCS High Pressure Safety Injection subsystem when entering MODE 4.

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION 3.5.3.a L02 A. Required HPSI train inoperable.	A.1 Restore required HPSI train to OPERABLE status.	1 hour
ACTION 3.5.3.a L03 B. Required Action and associated Completion Time not met.	B.1 Be in MODE ~~5~~ 4 with RCS cold leg temperature ≤ 250°F. (4)	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.5.3.1 SR 3.5.3.1 For required HPSI train, applicable portion of the ~~The~~ following SRs are applicable: (3) ~~[~~SR 3.5.2.1~~]~~ SR 3.5.2.6 SR 3.5.2.2 SR 3.5.2.7 ~~[~~SR 3.5.2.3~~]~~ ~~[~~SR 3.5.2.9~~]~~ SR 3.5.2.4 (2)	In accordance with applicable SRs

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3

LCO 3.5.3 One high pressure safety injection (HPSI) train shall be OPERABLE.

3.5.3 Applicability

APPLICABILITY: MODE 3 with pressurizer pressure < ~~[1700]~~ 1750 psia, MODE 4. (2)

ACTIONS

----------NOTE----------

ACTION 3.5.3.c

LCO 3.0.4.b is not applicable to ECCS High Pressure Safety Injection subsystem when entering MODE 4.

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION 3.5.3.a A. Required HPSI train inoperable.	A.1 Restore required HPSI train to OPERABLE status.	1 hour
ACTION 3.5.3.a L03 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 5.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.5.3 SR 3.5.3.1 For required HPSI train, applicable portion of the ~~The~~ following SRs are applicable: ~~[~~SR 3.5.2.1~~]~~ SR 3.5.2.6 SR 3.5.2.2 SR 3.5.2.7 ~~[~~SR 3.5.2.3~~]~~ ~~[~~SR 3.5.2.9~~]~~ SR 3.5.2.4	In accordance with applicable SRs (3) (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.3, ECCS - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Clarification is added to ISTS SR 3.5.3.1. The change clarifies the intent that only the required HPSI train portions of the listed SRs are applicable. This change is similar to the SR wording of SR 3.5.3.1 of ISTS NUREGs 1430 (Babcock and Wilcox) and 1431 (Westinghouse).

4. **Unit 1 only**: To support the Unit 1 low temperature overpressure protection (LTOP) analysis, all HPSI pumps must be incapable of injecting into the Reactor Coolant System (RCS) when RCS cold leg temperature is ≤ 236°F in MODE 4. Therefore, the change to the Applicability in the ITS to MODE 4 with RCS cold leg temperature > 250°F is made consistent with the # footnote in CTS to ensure both HPSI subsystems are blocked prior to RCS cold leg temperature dropping below 236°F. Core cooling requirements in MODE 4 when RCS cold leg temperature is ≤ 250°F are addressed by LCO 3.4.6, “RCS Loops – MODE 4.” Required Action B.1 is modified in the ITS to require action to exit the Applicability consistent with the change to the Applicability.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

Thus, only the SI and recirculation actuation manual channels and associated actuation logic to the required HPSI train are required in MODE 3 with pressurizer pressure < 1750 psia and MODE 4 with RCS cold leg temperature > 250°F.

BASES

BACKGROUND

, as it describes the design of the ECCS,

The Background section for Bases B 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

1750 when RCS cold leg temperature is > 250°F

In MODE 3 with pressurizer pressure < ~~1700~~ psia and in MODE 4, an ECCS train is defined as one high pressure safety injection (HPSI) subsystem. The HPSI ~~flow path~~ consists of piping, valves, and pumps that enable water from the refueling water tank (RWT) and the containment sump to be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

train

APPLICABLE SAFETY ANALYSES

~~The Applicable Safety Analyses section of Bases 3.5.2 is applicable to these Bases.~~

Due to the stable conditions associated with operation in MODE 4, and the reduced probability of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. Included in these reductions is that certain automatic safety injection (SI) actuation signals are not available. ~~In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.~~

Only one train of ECCS is required for ~~MODE 4~~. Protection against single failures is not relied on for this MODE of operation.

MODE 3 with pressurizer pressure < 1750 psia and MODE 4 with RCS cold leg temperature > 250°F.

ECCS - Shutdown satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

1750

In MODE 3 with pressurizer pressure < ~~1700~~ psia, an ECCS subsystem is composed of a single HPSI subsystem. Each HPSI subsystem includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT and transferring suction to the containment sump.

and MODE 4 when RCS cold leg temperature is > 250°F

During an event requiring ECCS actuation, a flow path is required to supply water from the RWT to the RCS via the HPSI pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs. Management of gas voids is important to ECCS OPERABILITY.

Plant licensing basis does not require an analysis to determine the effects of a loss of coolant accident (LOCA) or demonstration of ECCS equipment capability to mitigate a LOCA in MODE 3 with pressurizer pressure < 1750 psia or MODE 4 when RCS cold leg temperature is > 250°F. However, one HPSI subsystem is required to be OPERABLE in MODE 3 with pressurizer pressure < 1750 psia and MODE 4 with RCS cold leg temperature > 250°F to ensure sufficient ECCS flow is available to the core and adequate core cooling is maintained following a loss of RCS inventory.

BASES

when RCS cold leg temperature is ≤ 300°F

LCO (continued)

1750

With RCS pressure < ~~1700~~ psia, one HPSI pump is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements. The low pressure safety injection (LPSI) pumps may therefore be released from the ECCS train for use in shutdown cooling (SDC). In MODE 4 ~~with RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR~~, a maximum of one HPSI pump is allowed to be OPERABLE in accordance with LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

In addition, no HPSI pump can be OPERABLE when RCS cold leg temperature is ≤ 236°F in accordance with LCO 3.4.12.

3

3

3

APPLICABILITY

1750

In MODES 1, 2, and 3 with RCS pressure ≥ ~~1700~~ psia, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

1750

with RCS cold leg temperature > 250°F

In MODE 3 with RCS pressure < ~~1700~~ psia and in MODE 4, one OPERABLE ECCS train is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements.

3

3

MODE 4 when RCS cold leg temperature is ≤ 250°F and in

MODE 4 when RCS cold leg temperature is ≤ 250°F are addressed by LCO 3.4.6, "RCS Loops – MODE 4. Core cooling requirements in

In MODES 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

1

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable ECCS High Pressure Safety Injection subsystem. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS High Pressure Safety Injection subsystem and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With no HPSI pump OPERABLE, the unit is not prepared to respond to a loss of coolant accident. The 1 hour Completion Time to restore at least one HPSI train to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity or to initiate actions to place the unit in ~~MODE 5,~~ where an ECCS train is not required.

a condition

1

BASES

ACTIONS (continued)

B.1

MODE 4 with RCS cold leg temperature ≤ 250°F

When the Required Action cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is reasonable, based on operating experience, to reach ~~MODE 5~~ in an orderly manner and without challenging plant systems.

3

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

~~The applicable references from Bases 3.5.2 apply~~.

None

1

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

Thus, only the SI and recirculation actuation manual channels and associated actuation logic to the required HPSI train are required in MODE 3 with pressurizer pressure < 1750 psia and MODE 4.

BASES

BACKGROUND

, as it describes the design of the ECCS,

The Background section for Bases B 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

1750

In MODE 3 with pressurizer pressure < ~~1700~~ psia and in MODE 4, an ECCS train is defined as one high pressure safety injection (HPSI) subsystem. The HPSI ~~flow path~~ consists of piping, valves, and pumps that enable water from the refueling water tank (RWT) and the containment sump to be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

train

APPLICABLE SAFETY ANALYSES

~~The Applicable Safety Analyses section of Bases 3.5.2 is applicable to these Bases.~~

Due to the stable conditions associated with operation in MODE 4, and the reduced probability of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. Included in these reductions is that certain automatic safety injection (SI) actuation signals are not available. ~~In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.~~

Only one train of ECCS is required for ~~MODE 4~~. Protection against single failures is not relied on for this MODE of operation.

MODE 3 with pressurizer pressure < 1750 psia and MODE 4.

ECCS - Shutdown satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

1750

and MODE 4

LCO

In MODE 3 with pressurizer pressure < ~~1700~~ psia, an ECCS subsystem is composed of a single HPSI subsystem. Each HPSI subsystem includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to supply water from the RWT to the RCS via the HPSI pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs. Management of gas voids is important to ECCS OPERABILITY.

Plant licensing basis does not require an analysis to determine the effects of a loss of coolant accident (LOCA) or demonstration of ECCS equipment capability to mitigate a LOCA in MODE 3 with pressurizer pressure < 1750 psia or MODE 4. However, one HPSI subsystem is required to be OPERABLE in MODE 3 with pressurizer pressure < 1750 psia and MODE 4 to ensure sufficient ECCS flow is available to the core and adequate core cooling is maintained following a loss of RCS inventory.

BASES

when any RCS cold leg temperature is ≤ 240°F following entry from MODE 3 and when any RCS cold leg temperature is ≤ 252°F following entry from MODE 5, 3

LCO (continued)

1750

With RCS pressure < ~~1700~~ psia, one HPSI pump is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements. The low pressure safety injection (LPSI) pumps may therefore be released from the ECCS train for use in shutdown cooling (SDC). In MODE 4 ~~with RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR~~, a maximum of one HPSI pump is allowed to be OPERABLE in accordance with LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." 3

APPLICABILITY

1750

In MODES 1, 2, and 3 with RCS pressure ≥ ~~1700~~ psia, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2. 3

1750

In MODE 3 with RCS pressure < ~~1700~~ psia and in MODE 4, one OPERABLE ECCS train is acceptable without single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements. 3

In MODES 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable ECCS High Pressure Safety Injection subsystem. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS High Pressure Safety Injection subsystem and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With no HPSI pump OPERABLE, the unit is not prepared to respond to a loss of coolant accident. The 1 hour Completion Time to restore at least one HPSI train to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity or to initiate actions to place the unit in MODE 5, where an ECCS train is not required.

BASES

ACTIONS (continued)

B.1

When the Required Action cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

~~The applicable references from Bases 3.5.2 apply~~.

None

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.3, BASES, ECCS - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. Editorial change is made to the Background section of the ISTS Bases to clarify what constitutes a HPSI train since LCO 3.5.3 requires one HPSI "train" to be OPERABLE. In addition, the text added to the Applicable Safety Analysis section of the ISTS Bases clarifies that only the manual channel as associated actuation logic to the required HPSI train are required in MODE 4 when the Specification is applicable.

3. Changes have been made to be consistent with changes made to the Specifications.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.5.3, ECCS - SHUTDOWN

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 4

ITS 3.5.4, Refueling Water Tank (RWT)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

LCO 3.5.4 3.5.4 The refueling water tank shall be OPERABLE ~~with~~:

SR 3.5.4.2 ~~a.~~ ~~A minimum contained~~ volume 477,360 gallons ~~of borated water,~~

SR 3.5.4.3 ~~b.~~ ~~A minimum~~ boron concentration ~~of~~ 1900 ppm~~,~~

SR 3.5.4.1 ~~c.~~ ~~A maximum water temperature of~~ 100°F~~,~~

SR 3.5.4.1 ~~d.~~ ~~A minimum~~ water temperature ~~of~~ 55°F ~~when in MODES 1 and 2, and~~

SR 3.5.4.1 ~~e.~~ ~~A minimum water temperature of 40°F when in MODES 3 and 4~~

Verify RWT borated water

is ≥

Verify RWT

is ≥

and ≤ 2200 ppm

Verify RWT borated

is ≥

and ≤

M01

M02

Applicability **APPLICABILITY**: MODES 1, 2, 3 and 4.

Add proposed Action A

Add proposed Action C

Add proposed Action B

L01

L02

ACTION:

for reasons other than Condition A or C

RWT

ACTION D
ACTION E ~~With the~~ refueling water tank inoperable~~,~~ restore ~~the tank~~ to OPERABLE status within 1 hour or be in ~~at least HOT STANDBY~~ within 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

MODE 3

MODE 5

36

SURVEILLANCE REQUIREMENTS

~~4.5.4~~ ~~The RWT shall be demonstrated OPERABLE:~~

SR 3.5.4.2, SR 3.5.4.3 Frequency ~~a.~~ In accordance with the Surveillance Frequency Control Program ~~by:~~

volume

SR 3.5.4.2 ~~1.~~ Verify~~ing the~~ water ~~level~~ in the tank, and

SR 3.5.4.3 ~~2.~~ Verify~~ing the~~ boron concentration of the water.

SR 3.5.4.1 Frequency
SR 3.5.4.1 ~~b.~~ In accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing the~~ RWT temperature.

SR 3.5.4.1 Note NOTE Only required to be performed when ambient air temperature is < 55°F or > 100°F.

L03

A01

EMERGENCY CORE COOLING SYSTEMS

~~3/4.5.4~~ REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

LCO 3.5.4 3.5.4 The refueling water tank shall be OPERABLE ~~with~~:

SR 3.5.4.2 ~~a.~~ ~~A minimum contained~~ borated water volume (Verify RWT) 477,360 gallons (is ≥)~~,~~

SR 3.5.4.3 ~~b.~~ ~~A~~ boron concentration ~~of between~~ 1900 and 2200 ppm (Verify RWT) (is ≥) (ppm) (≤) ~~of boron, and~~

SR 3.5.4.1 ~~c~~. ~~A solution~~ temperature ~~of between~~ 55°F and 100°F. (Verify RWT borated water) (is ≥) (≤)

Applicability **APPLICABILITY**: MODES 1, 2, 3 and 4.

ACTION:

(Add proposed Action A) (Add proposed Action C) (Add proposed Action B) L01 L02

ACTION D ~~With the~~ refueling water tank inoperable~~,~~ restore ~~the tank~~ to OPERABLE status (for reasons other than Condition A or C) (RWT)

ACTION E within 1 hour or be in ~~at least~~ ~~HOT STANDBY~~ within 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ 36 hours. (MODE 3) (MODE 5)

SURVEILLANCE REQUIREMENTS

~~4.5.4~~ ~~The RWT shall be demonstrated OPERABLE:~~

SR 3.5.4.2, SR 3.5.4.3 Frequency ~~a.~~ In accordance with the Surveillance Frequency Control Program ~~by:~~

SR 3.5.4.2 ~~1.~~ Verify~~ing the~~ contained borated water volume in the tank, and

SR 3.5.4.3 ~~2.~~ Verify~~ing the~~ boron concentration of the water. (In accordance with the Surveillance Frequency Control Program) LA01

SR 3.5.4.1 and Frequency ~~b.~~ ~~At least once per 24 hours~~ ~~by~~ verify~~ing the~~ RWT temperature when

SR 3.5.4.1 NOTE ~~the outside~~ air temperature is ~~less then~~ 55°F or ~~greater than~~ 100°F. (ambient) (<) (>) (NOTE Only required to be performed) A01

DISCUSSION OF CHANGES
ITS 3.5.4, REFUELING WATER TANK(RWT)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS 3.5.4.b requires a minimum RWT boron concentration of 1900 ppm but does not require a maximum boron concentration. ITS SR 3.5.4.3 requires verification that RWT boron concentration is within a minimum and maximum limit. Similarly, Unit 2 CTS 3.5.4.b requires boron concentration within a minimum 1900 ppm and a maximum 2200 ppm. This changes the Unit 1 CTS by adding a RWT boron concentration maximum limit of 2200 ppm.

The purpose of CTS 3.5.4.b is to provide a RWT minimum boron concentration limit to ensure that, following a LOCA with a minimum level in the RWT, the reactor will remain subcritical in the cold condition following mixing of the RWT and RCS water volumes. The RWT maximum boron concentration limit is based on boron precipitation in the core following a LOCA which could result in reduced heat removal capability between the fuel cladding and the reactor coolant. This change is acceptable, because the maximum boron concentration limit provides additional assurance that the RWT is capable of performing its function and not impeding other mitigation systems from performing their safety functions.

This change is designated as more restrictive because it adds a RWT maximum boron concentration limit to the CTS.

M02 **Unit 1 only:** CTS 3.5.4.d requires a minimum RWT water temperature of 55°F when in MODES 1 and 2. CTS 3.5.4.e requires a minimum RWT water temperature of 40°F when in MODES 3 and 4. ITS SR 3.5.4.1 requires verification that RWT boron concentration is within a minimum and maximum limit, with the minimum limit the same for MODES 1, 2, 3, and 4. Similarly, Unit 2 CTS 3.5.4.c requires water temperature of between 55°F and 100°F in MODES 1, 2, 3 and 4. This changes the Unit 1 CTS by changing the RWT minimum water temperature to 55°F in MODES 3 and 4.

The purpose of CTS 3.5.4.e is to provide a RWT minimum water temperature in MODES 3 and 4 to ensure that the RWT is capable of performing its function when required. This change is acceptable because raising the minimum required water temperature from 40°F to 55°F in MODES 3 and 4 provides additional margin to the low temperature design limit and provides assurance that the RWT will be capable of performing its function. In addition, raising the minimum temperature of the RWT will further minimize the reactivity impact as a result of

an inadvertent ECCS injection while shutdown. This change is designated as more restrictive because it changes the RWT minimum water temperature in MODES 3 and 4 to a higher temperature than required in CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 **Unit 2 only:** (*Type 6 – Removal of SR Frequency to the Surveillance Frequency Control Program*) CTS 4.5.4.b requires verification that RWT borated water temperature is within limits at least once per 24 hours when the outside (i.e., ambient) air temperature is less than 55°F or greater than 100°F. ITS SR 3.5.4.1 requires a similar Surveillance and specifies a periodic Frequency of, "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified periodic Frequency to the Surveillance Frequency Control Program.

The removal of periodic Surveillance Requirement Frequencies is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to periodically verify RWT borated water temperature is within limits. The existing Surveillance Frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. The control of changes to periodic Surveillance Frequencies is in accordance with the Surveillance Frequency Control Program requirements specified in CTS 6.8.4 (ITS 5.5.16). This program ensures that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met. This change is designated as a less restrictive removal of detail change because Surveillance Frequencies are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 – Relaxation of Completion Time)* CTS 3.5.4 ACTION allows 1 hour to restore an inoperable RWT. ITS 3.5.4 ACTION A allows 8 hours to restore the RWT to OPERABLE status if the inoperability is due to the RWT boron concentration not within limits. ITS 3.5.4 ACTION C allows 8 hours to restore the RWT to OPERABLE status if the inoperability is due to the RWT borated water temperature not within limits. ITS 3.5.4 ACTION D requires the restoration of the RWT to an OPERABLE status within 1 hour for reasons other than Condition A or Condition C (e.g., RWT water volume not within limits). This changes the CTS by expanding the Completion Time for restoration of an inoperable RWT due to boron concentration or borated water temperature not within limits from 1 hour to 8 hours.

The purpose of CTS 3.5.4 Action is to require prompt action to restore the RWT to OPERABLE status. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering a reasonable time to restore RWT parameters and the low probability of a DBA occurring during the allowed Completion Time. The primary function of the RWT is to contain sufficient borated water volume to support the ECCS during the injection phase following a LOCA. This RWT water volume continues to be available while in the Condition with the RWT boron concentration or the RWT borated water temperature not within limits. As a result, the most important safety function of the RWT can still be provided. The allowed Completion Time of 8 hours to restore the RWT to within limits was developed considering the time required to change the boron concentration or water temperature and that the contents of the tank are still available for injection. Given the remaining abilities of the RWT, requiring a plant shutdown after one hour is not warranted. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.5.4 ACTION, states, "With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours."

ITS 3.5.4 ACTION A allows 8 hours to restore the RWT to OPERABLE status if the inoperability is due to the RWT boron concentration not within limits. ITS 3.5.4 ACTION C allows 8 hours to restore the RWT to OPERABLE status if the inoperability is due to the RWT borated water temperature not within limits. ITS 3.5.4 ACTION D requires the restoration of the RWT to an OPERABLE status within 1 hour for reasons other than Condition A or Condition C (e.g., RWT water volume not within limits). ITS ACTION E requires that with the Required Action and associated Completion Time of Condition C (borated water temperature) or D (borated water volume) not met, then be in MODE 3 within 6 hours and MODE 5 within 36 hours consistent with CTS. In addition, the end state of ITS 3.5.4 is MODE 5 consistent with CTS.

ITS ACTION B requires that with the Required Action and associated Completion Time of Condition A (boron concentration) not met, then be in MODE 3 within 6 hours and includes a NOTE modifying Required Action B.1 that states, “LCO 3.0.4.a is not applicable when entering MODE 3 or MODE 4.” This changes CTS by changing the end state of CTS 3.5.4.b, RWT boron concentration, from MODE 5 (Cold Shutdown) to MODE 3 (Hot Standby).

This change is acceptable because the modification of the end state from MODE 5 to MODE 3, with respect to RWT boron concentration, is consistent with CE NPSD-1186-A, Revision 00, “Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs,” dated October 2001 (ADAMS Accession No. ML110410539) and Technical Specifications Task Force (TSTF) traveler TSTF- 422, Revision 2, “Change in Technical Specifications End States, (CE NPSD-1186),“ dated December 22, 2009 (ADAMS Accession No. ML103270197) allowances.

Florida Power & Light Company provided the regulatory commitments regarding the implementation of TSTF-422, Revision 2, in License Amendment Request for Adoption of Technical Specifications Task Force (TSTF) Traveler TSTF-422, Revision 2, "Change in Technical Specifications End States, (CE NPSD-1186)," Using the Consolidated Line Item Improvement Process (ADAMS Accession No. ML16118A272). The NRC issued Amendment Nos. 234 and 184 to Renewed Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2, respectively (ADAMS Accession No. ML16210A374).

This change is designated as less restrictive because instead of requiring the plant to achieve COLD SHUTDOWN (MODE 5) with the RWT boron concentration not restored to within limits, the ITS end state is HOT STANDBY (MODE 3) with the Note that LCO 3.0.4.a not applicable when entering MODE 3 or MODE 4.

L03 **Unit 1 only:** (*Category 7 – Relaxation of Surveillance Frequency)* CTS 4.5.4.b states that the RWT be demonstrated OPERABLE by verifying the RWT water temperature within limits in accordance with the Surveillance Frequency Control Program. ITS SR 3.5.4.1 also states that the RWT be demonstrated OPERABLE by verifying the RWT water temperature within limits in accordance with the Surveillance Frequency Control Program. ITS SR 3.5.4.1 is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating temperature limits of the RWT. This changes the CTS by adopting the ITS SR 3.5.4.1 Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating temperature limits of the RWT.

The purpose of CTS 4.5.4.b and ITS SR 3.5.4.1 is to verify RWT water temperature within limits. This change is acceptable because with ambient temperatures within the RWT operating temperature limits, the RWT temperature should not exceed the limits since the RWT is exposed to ambient temperature and is not insulated. Because of the RWT design, tank temperature tracks closely with outside ambient temperature. Thus, periodic verification of the RWT temperature is unnecessary while ambient temperature is between 55°F and 100°F. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Tank (RWT)

LCO 3.5.4

LCO 3.5.4 The RWT shall be OPERABLE.

3.5.4 Applicability

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.5.4 ACTION DOC L01 A. RWT boron concentration not within limits.	A.1 Restore RWT to OPERABLE status.	8 hours
3.5.4 ACTION DOC L02 B. Required Action and associated Completion Time of Condition A not met.	B.1 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 3 or MODE 4. ---------- Be in MODE 3.	6 hours
3.5.4 ACTION DOC L01 C. RWT borated water temperature not within limits.	C.1 Restore RWT to OPERABLE status.	8 hours
3.5.4 ACTION D. RWT inoperable for reasons other than Condition A or C.	D.1 Restore RWT to OPERABLE status	1 hour
3.5.4 ACTION E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.4.1	----------NOTE---------- [Only required to be performed when ambient air temperature is < [40]°F or > [100]°F.]	
SR 3.5.4.1	Verify RWT borated water temperature is ≥ [40]°F and ≤ [100]°F.	[24 hours OR In accordance with the Surveillance Frequency Control Program]
SR 3.5.4.2	Verify RWT borated water volume is ≥ [362,800 gallons, (88)%] [above the ECCS suction connection].	[7 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.5.4.3	Verify RWT boron concentration is ≥ [1720] ppm and ≤ [2500] ppm.	[7 days OR In accordance with the Surveillance Frequency Control Program]

DOC L03

LCO 3.5.4.c,d
SR 4.5.4.b
DOC M02

LCO 3.5.4.a
SR 4.5.4.a.1

LCO 3.5.4.b
SR 4.5.4.a.2
DOC M01

55

55

477,360

1900

2200

2

2

2

2

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Tank (RWT)

LCO 3.5.4 The RWT shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.5.4 ACTION DOC L01	A. RWT boron concentration not within limits.	A.1 Restore RWT to OPERABLE status.	8 hours
3.5.4 ACTION DOC L02	B. Required Action and associated Completion Time of Condition A not met.	B.1 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 3 or MODE 4. ---------- Be in MODE 3.	6 hours
3.5.4 ACTION DOC L01	C. RWT borated water temperature not within limits.	C.1 Restore RWT to OPERABLE status.	8 hours
3.5.4 ACTION	D. RWT inoperable for reasons other than Condition A or C.	D.1 Restore RWT to OPERABLE status	1 hour
3.5.4 ACTION	E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.4.1 LCO 3.5.4.c SR 4.5.4.b	----------NOTE---------- [Only required to be performed when ambient air temperature is < [40]°F or > [100]°F.] ---------- (55) ---------- Verify RWT borated water temperature is ≥ [40]°F and ≤ [100]°F. (55)	[24 hours OR In accordance with the Surveillance Frequency Control Program] (2)
SR 3.5.4.2 LCO 3.5.4.a SR 4.5.4.a.1	Verify RWT borated water volume is ≥ [362,800 gallons, (88)%] [above the ECCS suction connection]. (477,360)	[7 days OR In accordance with the Surveillance Frequency Control Program] (2)
SR 3.5.4.3 LCO 3.5.4.b SR 4.5.4.a.2	Verify RWT boron concentration is ≥ [1720] (1900) ppm and ≤ [2500] (2200) ppm.	[7 days OR In accordance with the Surveillance Frequency Control Program] (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.4, REFUELING WATER TANK (RWT)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Tank (RWT)

BASES

BACKGROUND

The RWT supports the ECCS and the Containment Spray System by providing a source of borated water for Engineered Safety Feature (ESF) pump operation.

The RWT supplies two ECCS trains by separate, redundant supply headers. Each header also supplies one train of the Containment Spray System. A motor operated isolation valve is provided in each header to allow the operator to isolate the usable volume of the RWT from the ECCS after the ESF pump suction has been transferred to the containment sump following depletion of the RWT during a loss of coolant accident (LOCA). A separate header is used to supply the Chemical and Volume Control System (CVCS) from the RWT. Use of a single RWT to supply both trains of the ECCS is acceptable since the RWT is a passive component, and passive failures are not assumed to occur coincidently with the Design Basis Event during the injection phase of an accident. Not all the water stored in the RWT is available for injection following a LOCA; the location of the ECCS suction piping in the RWT will result in some portion of the stored volume being unavailable.

The high pressure safety injection (HPSI), low pressure safety injection (LPSI), and containment spray pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at shutoff head conditions. These lines discharge back to the RWT, which vents to the atmosphere. When the suction for the HPSI and containment spray pumps is transferred to the containment sump, this flow path must be isolated to prevent a release of the containment sump contents to the RWT. If not isolated, this flow path could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ESF pumps.

This LCO ensures that:

a. The RWT contains sufficient borated water to support the ECCS during the injection phase,

b. Sufficient water volume exists in the containment sump to support continued operation of the ESF pumps at the time of transfer to the recirculation mode of cooling, and

c. The reactor remains subcritical following a LOCA.

BASES

BACKGROUND (continued)

Insufficient water inventory in the RWT could result in insufficient cooling capacity of the ECCS when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWT provides a source of borated water to the HPSI, LPSI, containment spray, and charging pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of Bases B 3.5.2, "ECCS - Operating," and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWT in order to evaluate their effects in relation to the acceptance limits.

The volume limit of ~~[362,800]~~ 477,360 gallons is based on two factors: 2

a. Sufficient deliverable volume must be available to provide at least 20 minutes ~~(plus a 10% margin)~~ of full flow from all ESF pumps prior to reaching a low level switchover to the containment sump for recirculation and 1

b. The containment sump water volume must be sufficient to support continued ESF pump operation after the switchover to recirculation occurs. This sump volume water inventory is supplied by the RWT borated water inventory.

Twenty minutes is the point at which 75% of the design flow of one HPSI pump is capable of meeting or exceeding the decay heat boiloff rate.

When ESF pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head (NPSH) for the HPSI and containment spray pumps. The RWT capacity must be sufficient to supply this amount of water without considering the inventory added from the safety injection tanks or Reactor Coolant System (RCS), but accounting for loss of inventory to containment subcompartments and reservoirs due to containment spray operation and to areas outside containment due to leakage from ECCS injection and recirculation equipment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

1900

The ~~[1720]~~ ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum level in the RWT, the reactor will remain subcritical in the cold condition following mixing of the RWT and RCS water volumes. Small break LOCAs assume that all control rods are inserted, except for the control element assembly (CEA) of highest worth, which is withdrawn from the core. Large break LOCAs assume that all CEAs remain withdrawn from the core. The most limiting case occurs at beginning of core life. 2

2200

The maximum boron limit of ~~[2500]~~ ppm in the RWT is based on boron precipitation in the core following a LOCA. With the reactor vessel at saturated conditions, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point will be reached where boron precipitation will occur in the core. Post LOCA emergency procedures direct the operator to establish simultaneous hot and cold leg injection to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA. Boron concentrations in the RWT in excess of the limit could result in precipitation earlier than assumed in the analysis. 2

55

The upper limit of ~~[~~100~~]~~°F and the lower limit of ~~[40]~~°F on RWT temperature are the limits assumed in the accident analysis. Although RWT temperature affects the outcome of several analyses, the upper and lower limits established by the LCO are not limited by any of these analyses. 2

The RWT satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The RWT ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA) and to cool and cover the core in the event of a LOCA, that the reactor remains subcritical following a DBA, and that an adequate level exists in the containment sump to support ESF pump operation in the recirculation mode.

To be considered OPERABLE, the RWT must meet the limits established in the SRs for water volume, boron concentration, and temperature.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the RWT OPERABILITY requirements are dictated by the ECCS and Containment Spray System OPERABILITY requirements. Since ~~both~~ the ECCS ~~and the Containment Spray System~~ must be OPERABLE in MODES 1, 2, 3, and 4, the RWT must be OPERABLE to support ~~their~~ its operation. 1 1

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With RWT boron concentration not within limits, it must be returned to within limits within 8 hours. In this condition neither the ECCS nor the Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE condition. The allowed Completion Time of 8 hours to restore the RWT to within limits was developed considering the time required to change boron concentration and that the contents of the tank are still available for injection.

B.1

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~ 3

If the RWT boron concentration is not restored to within limits within the associated Completion Time, the plant must be brought to MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant

BASES

ACTIONS (continued)

systems. Reference 2 demonstrated that is it acceptable to remain in MODE 3 in this condition because the boron concentration limit is based on MODE 1 events which are unlikely in MODE 3, such as a LOCA, and conditions which do not exist in MODE 3, such as a critical core with all rods out. Since the anticipated deviations from the RWT boron concentration requirements are expected to be small and the ability to correct the deficiency is expected to be readily available, entry into MODE 4 or 5 is not necessary.

Required Action B.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3 or MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 or MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

C.1

With RWT water temperature not within limits, it must be returned to within limits within 8 hours. In this condition neither the ECCS nor the Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE condition. The allowed Completion Time of 8 hours to restore the RWT to within limits was developed considering the time required to change water temperature and that the contents of the tank are still available for injection.

D.1 or RWT inoperable for reasons other than boron concentration, borated water temperature, or borated water volume, 1

With RWT borated water volume not within limits, it must be returned to within limits within 1 hour. In this condition, neither the ECCS nor Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the unit in a MODE in which these systems are not required. The allowed Completion Time of 1 hour to restore the RWT to OPERABLE status is based on this condition simultaneously affecting multiple redundant trains.

BASES

ACTIONS (continued)

E.1 and E.2

If the RWT temperature or water volume cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

RWT borated water temperature shall be verified to be within the limits assumed in the accident analysis. ~~[This Frequency of 24 hours has been shown to be sufficient to identify temperature changes that approach either acceptable limit.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating temperature limits of the RWT. With ambient temperatures within this range, the RWT temperature should not exceed the limits.

SR 3.5.4.2

Above minimum RWT water volume level shall be verified. This Frequency ensures that a sufficient initial water supply is available for injection and to support continued ESF pump operation on recirculation. ~~[Since the RWT volume is normally stable and is provided with a Low Level Alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.~~ 2

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

SR 3.5.4.3

Boron concentration of the RWT shall be verified to be within the required range. This Frequency ensures that the reactor will remain subcritical following a LOCA. Further, it ensures that the resulting sump pH will be maintained in an acceptable range such that boron precipitation in the core will not occur earlier than predicted and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. ~~[Since the RWT volume is normally stable, a 7 day sampling Frequency is appropriate and has been shown through operating experience to be acceptable.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

REFERENCES

1. FSAR, ~~Chapter~~ ~~[6]~~ and ~~Chapter~~ ~~[15]~~. (U; Section 6.2; Section 15.6) 1 2
2. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Tank (RWT)

BASES

BACKGROUND

The RWT supports the ECCS and the Containment Spray System by providing a source of borated water for Engineered Safety Feature (ESF) pump operation.

The RWT supplies two ECCS trains by separate, redundant supply headers. Each header also supplies one train of the Containment Spray System. A motor operated isolation valve is provided in each header to allow the operator to isolate the usable volume of the RWT from the ECCS after the ESF pump suction has been transferred to the containment sump following depletion of the RWT during a loss of coolant accident (LOCA). A separate header is used to supply the Chemical and Volume Control System (CVCS) from the RWT. Use of a single RWT to supply both trains of the ECCS is acceptable since the RWT is a passive component, and passive failures are not assumed to occur coincidently with the Design Basis Event during the injection phase of an accident. Not all the water stored in the RWT is available for injection following a LOCA; the location of the ECCS suction piping in the RWT will result in some portion of the stored volume being unavailable.

The high pressure safety injection (HPSI), low pressure safety injection (LPSI), and containment spray pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at shutoff head conditions. These lines discharge back to the RWT, which vents to the atmosphere. When the suction for the HPSI and containment spray pumps is transferred to the containment sump, this flow path must be isolated to prevent a release of the containment sump contents to the RWT. If not isolated, this flow path could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ESF pumps.

This LCO ensures that:

a. The RWT contains sufficient borated water to support the ECCS during the injection phase,

b. Sufficient water volume exists in the containment sump to support continued operation of the ESF pumps at the time of transfer to the recirculation mode of cooling, and

c. The reactor remains subcritical following a LOCA.

BASES

BACKGROUND (continued)

Insufficient water inventory in the RWT could result in insufficient cooling capacity of the ECCS when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWT provides a source of borated water to the HPSI, LPSI, containment spray, and charging pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of Bases B 3.5.2, "ECCS - Operating," and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWT in order to evaluate their effects in relation to the acceptance limits.

The volume limit of ~~[362,800]~~ 477,360 gallons is based on two factors: 2

a. Sufficient deliverable volume must be available to provide at least 20 minutes ~~(plus a 10% margin)~~ of full flow from all ESF pumps prior to reaching a low level switchover to the containment sump for recirculation and 1

b. The containment sump water volume must be sufficient to support continued ESF pump operation after the switchover to recirculation occurs. This sump volume water inventory is supplied by the RWT borated water inventory.

Twenty minutes is the point at which 75% of the design flow of one HPSI pump is capable of meeting or exceeding the decay heat boiloff rate.

When ESF pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head (NPSH) for the HPSI and containment spray pumps. The RWT capacity must be sufficient to supply this amount of water without considering the inventory added from the safety injection tanks or Reactor Coolant System (RCS), but accounting for loss of inventory to containment subcompartments and reservoirs due to containment spray operation and to areas outside containment due to leakage from ECCS injection and recirculation equipment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

1900

The ~~[1720]~~ ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum level in the RWT, the reactor will remain subcritical in the cold condition following mixing of the RWT and RCS water volumes. Small break LOCAs assume that all control rods are inserted, except for the control element assembly (CEA) of highest worth, which is withdrawn from the core. Large break LOCAs assume that all CEAs remain withdrawn from the core. The most limiting case occurs at beginning of core life. 2

2200

The maximum boron limit of ~~[2500]~~ ppm in the RWT is based on boron precipitation in the core following a LOCA. With the reactor vessel at saturated conditions, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point will be reached where boron precipitation will occur in the core. Post LOCA emergency procedures direct the operator to establish simultaneous hot and cold leg injection to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA. Boron concentrations in the RWT in excess of the limit could result in precipitation earlier than assumed in the analysis. 2

55

The upper limit of ~~[~~100~~]~~°F and the lower limit of ~~[40]~~°F on RWT temperature are the limits assumed in the accident analysis. Although RWT temperature affects the outcome of several analyses, the upper and lower limits established by the LCO are not limited by any of these analyses. 2

The RWT satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The RWT ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA) and to cool and cover the core in the event of a LOCA, that the reactor remains subcritical following a DBA, and that an adequate level exists in the containment sump to support ESF pump operation in the recirculation mode.

To be considered OPERABLE, the RWT must meet the limits established in the SRs for water volume, boron concentration, and temperature.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the RWT OPERABILITY requirements are dictated by the ECCS and Containment Spray System OPERABILITY requirements. Since ~~both~~ the ECCS ~~and the Containment Spray System~~ must be OPERABLE in MODES 1, 2, 3, and 4, the RWT must be OPERABLE to support ~~their~~ its operation. 1 1

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With RWT boron concentration not within limits, it must be returned to within limits within 8 hours. In this condition neither the ECCS nor the Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE condition. The allowed Completion Time of 8 hours to restore the RWT to within limits was developed considering the time required to change boron concentration and that the contents of the tank are still available for injection.

B.1

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~ 3

If the RWT boron concentration is not restored to within limits within the associated Completion Time, the plant must be brought to MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant

BASES

ACTIONS (continued)

systems. Reference 2 demonstrated that is it acceptable to remain in MODE 3 in this condition because the boron concentration limit is based on MODE 1 events which are unlikely in MODE 3, such as a LOCA, and conditions which do not exist in MODE 3, such as a critical core with all rods out. Since the anticipated deviations from the RWT boron concentration requirements are expected to be small and the ability to correct the deficiency is expected to be readily available, entry into MODE 4 or 5 is not necessary.

Required Action B.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3 or MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 or MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

C.1

With RWT water temperature not within limits, it must be returned to within limits within 8 hours. In this condition neither the ECCS nor the Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE condition. The allowed Completion Time of 8 hours to restore the RWT to within limits was developed considering the time required to change water temperature and that the contents of the tank are still available for injection.

D.1 or RWT inoperable for reasons other than boron concentration, borated water temperature, or borated water volume, 1

With RWT borated water volume not within limits, it must be returned to within limits within 1 hour. In this condition, neither the ECCS nor Containment Spray System can perform their design functions; therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the unit in a MODE in which these systems are not required. The allowed Completion Time of 1 hour to restore the RWT to OPERABLE status is based on this condition simultaneously affecting multiple redundant trains.

BASES

ACTIONS (continued)

E.1 and E.2

If the RWT temperature or water volume cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

RWT borated water temperature shall be verified to be within the limits assumed in the accident analysis. ~~[This Frequency of 24 hours has been shown to be sufficient to identify temperature changes that approach either acceptable limit.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~]~~

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating temperature limits of the RWT. With ambient temperatures within this range, the RWT temperature should not exceed the limits.

SR 3.5.4.2

Above minimum RWT water volume level shall be verified. This Frequency ensures that a sufficient initial water supply is available for injection and to support continued ESF pump operation on recirculation. ~~[Since the RWT volume is normally stable and is provided with a Low Level Alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.~~ 2

~~OR~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 2

SR 3.5.4.3

Boron concentration of the RWT shall be verified to be within the required range. This Frequency ensures that the reactor will remain subcritical following a LOCA. Further, it ensures that the resulting sump pH will be maintained in an acceptable range such that boron precipitation in the core will not occur earlier than predicted and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized~~. [Since the RWT volume is normally stable, a 7 day sampling Frequency is appropriate and has been shown through operating experience to be acceptable.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 2

REFERENCES

1. FSAR, ~~Chapter~~ ~~[6]~~ and ~~Chapter~~ ~~[15]~~. (U) Section 6.2 Section 15.6 1 2
2. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.4, BASES, REFUELING WATER TANK (RWT)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. The Reviewer's Note provides the NRC Reviewer information needed to meet the requirements. This Note is not meant to be retained in the final version of the plant specific submittal.

   CTS 3.5.4 ACTION, states, "With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours." ITS 3.5.4 ACTION A allows 8 hours to restore the RWT to OPERABLE status if the inoperability is due to the RWT boron concentration not within limits. ITS ACTION B requires that with the Required Action and associated Completion Time of Condition A (boron concentration) not met, then be in MODE 3 within 6 hours. The end state of ITS 3.5.4 is MODE 5. In addition, ITS ACTION B includes a NOTE modifying Required Action B.1 that states, "LCO 3.0.4.a is not applicable when entering MODE 3 or MODE 4." This changes CTS by modifying the end state of CTS 3.5.4.b, RWT boron concentration. The modification of the end state from MODE 5 to MODE 3, with respect to RWT boron concentration, is consistent with CE NPSD-1186-A, Revision 00, "Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs," dated October 2001 (ADAMS Accession No. ML110410539) and Technical Specifications Task Force (TSTF) traveler TSTF- 422, Revision 2, "Change in Technical Specifications End States, (CE NPSD-1186)," dated December 22, 2009 (ADAMS Accession No. ML103270197) allowances.

   FPL (PSL Unit 1 and Unit 2) provided the regulatory commitments regarding the implementation of TSTF-422, Revision 2, in License Amendment Request for Adoption of Technical Specifications Task Force (TSTF) Traveler TSTF-422, Revision 2, "Change in Technical Specifications End States, (CE NPSD-1186)," Using the Consolidated Line Item Improvement Process (ADAMS Accession No. ML16118A272). The NRC issued Amendment Nos. 234 and 184 to Renewed Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2, respectively (ADAMS Accession No. ML16210A374).

4. **Unit 2 Only:** The Reviewer's Note has been deleted. The Reviewer's Note provides the NRC Reviewer information needed to meet the requirements. This Note is not meant to be retained in the final version of the plant specific submittal. CTS 4.5.4.b requires verification that RWT borated water temperature is within limits at least once per 24 hours when the outside (i.e., ambient) air temperature is less than 55°F or greater than 100°F. The Surveillance Frequency Control Program was incorporated into the CTS by License Amendment 173, dated June 22, 2015 (ADAMS Accession No. 15127A066). The 24 hour Frequency specified in CTS 4.5.4.b is proposed to be relocated to the Surveillance Frequency Control Program (See Discussion of Change LA01).

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.5.4, REFUELING WATER TANK (RWT)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

ITS 3.5.5, Trisodium Phosphate (TSP) - Unit 2 only

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the following valves are in the indicated positions with power to the valve operators removed:

See ITS 3.5.2

Valve Number	Valve Function	Valve Position
a. V3733 V3734	a. SIT Vent Valves	a. Locked Closed
b. V3735 V3736	b. SIT Vent Valves	b. Locked Closed
c. V3737 V3738 V3739 V3740	c. SIT Vent Valves	c. Locked Closed

b. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.*

c. In accordance with the Surveillance Frequency Control Program by verifying ECCS locations susceptible to gas accumulation are sufficiently filled with water.

. d. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. At least once daily of the areas affected within containment by the containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

SR 3.5.5.1
SR 3.5.5.2
Frequency

~~e.~~ In accordance with the Surveillance Frequency Control Program ~~by:~~

A01

1. Verifying automatic isolation and interlock action of the shutdown cooling system from Reactor Coolant System when RCS pressure (actual or simulated) is greater than or equal to 515 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when RCS pressure (actual or simulated) is greater than or equal to 276 psia.

* Not required to be met for system vent flow paths opened under administrative control.

A01

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

See ITS 3.5.2

Add LCO 3.5.5

A02

LCO 3.5.5
SR 3.5.5.1

The TSP baskets shall contain ≥ 173 ft^3 of active TSP

Add ACTION A, B

L01

3. Verify~~ing that a minimum total of~~ 173 cubic feet of ~~solid granular trisodium phosphate dodecahydrate (~~TSP~~) is contained within~~ the TSP ~~storage~~ baskets~~.~~

contain

active

SR 3.5.5.2

4. Verify~~ing~~ that ~~when~~ a ~~representative~~ sample ~~of 70.5 ± 0.5 grams of TSP~~ from ~~a~~ TSP ~~storage~~ basket ~~is submerged, without agitation, in 10.0 ± 0.1 gallons of 120 ± 10°F borated water representative~~ of ~~the~~ RWT~~,~~ ~~the~~ pH ~~of the mixed solution is raised to greater than or equal to 7 within 4 hours~~.

the

water

LA01

provides adequate

adjustment

f. In accordance with the Surveillance Frequency Control Program, during shutdown, by:

1. Verifying that each automatic valve in the flow paths actuates to its correct position on SIAS and/or RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pumps.

   b. Low-Pressure Safety Injection pumps.

   c. Charging Pumps

3. Verifying that upon receipt of an actual or simulated Recirculation Actuation Signal: each low-pressure safety injection pump stops, each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes.

g. By verifying that each of the following pumps develops the specified total developed head when tested pursuant to the INSERVICE TESTING PROGRAM:

1. High-Pressure Safety Injection pumps.

2. Low-Pressure Safety Injection pumps.

h. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1. During valve stroking operation or following maintenance on the valve and prior to declaring the valve OPERABLE when the ECCS subsystems are required to be OPERABLE.

DISCUSSION OF CHANGES
ITS 3.5.5, TRISODIUM PHOSPHATE (TSP) – UNIT 2 ONLY

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.5.2 requires two ECCS subsystems to be OPERABLE with each subsystem comprised of specific components and flowpaths and is applicable in MODES 1, 2, and 3 with pressurizer pressure ≥ 1750 psia. CTS 4.5.2.e.3 and e.4 provide requirements for trisodium phosphate dodecahydrate (TSP). ITS LCO 3.5.5, "Trisodium Phosphate (TSP)," states that the TSP baskets shall contain ≥ 173 ft^3 of active TSP. The Applicability for this requirement is MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 1750 psia, consistent with the CTS Applicability. ITS ACTIONS are provided to restore TSP to within limits within 72 hours or place the plant in MODE 3 within 6 hours and reduce pressurizer pressure to < 1750 psia within 12 hours (See Discussion of Change L01 for change to CTS actions). This changes the CTS by providing a separate Specification for TSP requirements.

The purpose of CTS 4.5.2.e.3 and e.4 (TSP Surveillances) is to verify an adequate amount of active TSP is available to maintain pH in the containment sump during a LOCA to promote iodine retention in the sump during the recirculation phase of ECCS and containment spray, thereby limiting radiological consequences. This function is not related to the OPERABILITY of the ECCS trains and is more appropriately addressed by a separate Specification. This change is designated as administrative because providing a separate Specification for TSP requirements does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS SR 4.5.2.e.4 states that when a representative sample of 70.5 + 0.5 grams of TSP from a TSP storage basket is submerged, without agitation, in 10.0 ± 0.1 gallons of 120 ± 10°F borated water representative of the

RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours. ITS SR 3.5.5.2 requires verification that a sample from the TSP baskets provides adequate pH adjustment of RWT water, and does not retain this detail. This changes the CTS by relocating the details for verification of the TSP sample for pH adjustment.

The removal of these details, that are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS SR 3.5.5.2 retains the requirement to periodically verify a TSP sample for pH adjustment. The removed details that state "a representative sample of 70.5 + 0.5 grams of TSP from a TSP storage basket is submerged, without agitation, in 10.0 $\pm$ 0.1 gallons of 120 $\pm$ 10°F borated water representative of the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours," are moved to the ITS Bases. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled.

This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.5.2 does not contain ACTIONS to follow if the TSP Surveillances are not met. Therefore, if the TSP Surveillances are not met, CTS 3.0.3 would be required. CTS 3.0.3 requires action to be initiated within 1 hour and place the unit, as applicable, in hot standby (MODE 3) within the next 6 hours, hot shutdown (MODE 4) within the following 6 hours, and cold shutdown (MODE 5) within the subsequent 24 hours. Since the TSP Surveillances are applicable in MODES 1, 2, and 3 with pressurizer pressure ≥ 1750 psia, upon entry into CTS 3.0.3, the unit would have be placed in MODE 3 within 7 hours from entry and pressurizer pressure reduced below 1750 psia before MODE 4 is required to be reached within 13 hours from entry. ITS 3.5.5 contains ACTIONS to follow if the TSP LCO is not met. If the LCO is not met, 72 hours are provided to restore TSP to within limits. If this action is not completed within the 72 hours, the plant must be placed in MODE 3 within 6 hours and reduce pressurizer pressure to < 1750 psia within 12 hours. This changes the CTS by providing time to restore TSP to within limits before a plant shutdown is required.

The purpose of CTS 4.5.2.e.3 and e.4 (TSP Surveillances) is to verify an adequate amount of active TSP is available to maintain pH in the containment sump during a LOCA. During plant operation the containment sump is not accessible, and corrections may not be possible. However, if possible, this change provides time to restore TSP within limits rather than requiring a plant shutdown. If active TSP is not within the required limit, TSP must be restored

within the required limit in 72 hours. The pH control of the ECCS recirculation and containment spray solution for corrosion protection and iodine removal is reduced in this condition. This change is acceptable because the 72 hour Completion Time takes into account that the condition, which caused the TSP not within limits, would most likely allow the ECCS and Containment Spray System to continue to provide some capability for pH adjustment and iodine removal. The 72-hour Completion Time also considers that the Containment Spray System would continue to be available to remove some iodine from the containment atmosphere in the event of a DBA and the low probability of the worst case DBA occurring during this period. Therefore, allowing time to evaluate the TSP degradation and make any adjustments is acceptable. If TSP cannot be restored to within limits in 72 hours, the unit must be placed in a condition where the LCO no longer applies; MODE 3 with pressurizer pressure < 1750 psia. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Trisodium Phosphate (TSP)

4.5.2.e.3 LCO 3.5.5 The TSP baskets shall contain ≥ ~~[291]~~ 173 ft^3 of active TSP. 2

3.5.2 Applicability APPLICABILITY: ~~MODES 1, 2, and 3~~ MODE 1 and 2, MODE 3 with pressurizer pressure ≥ 1750 psia. 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 A. TSP not within limits.	A.1 Restore TSP to within limits.	72 hours
DOC L01 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ~~Be in MODE 4.~~ Reduce pressurizer pressure < 1750 psia.	6 hours ~~[~~12~~]~~ hours 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.5.2.e.3 SR 3.5.5.1 Verify the TSP baskets contain ≥ ~~[291]~~ 173 ft^3 of trisodium phosphate.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.e.4 SR 3.5.5.2	Verify that a sample from the TSP baskets provides adequate pH adjustment of RWT water.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.5, TRISODIUM PHOSPHATE (TSP) – UNIT 2 ONLY

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Trisodium Phosphate (TSP)

BASES

BACKGROUND

Trisodium phosphate (TSP) is placed on the floor or in the sump of the containment building to ensure that iodine, which may be dissolved in the recirculated reactor cooling water following a loss of coolant accident (LOCA), remains in solution. TSP also helps inhibit stress corrosion cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident.

Fuel that is damaged during a LOCA will release iodine in several chemical forms to the reactor coolant and to the containment atmosphere. A portion of the iodine in the containment atmosphere is washed to the sump by containment sprays. The emergency core cooling water is borated for reactivity control. This borated water causes the sump solution to be acidic. In a low pH (acidic) solution, dissolved iodine will be converted to a volatile form. The volatile iodine will evolve out of solution into the containment atmosphere, significantly increasing the levels of airborne iodine. The increased levels of airborne iodine in containment contribute to the radiological releases and increase the consequences from the accident due to containment atmosphere leakage.

After a LOCA, the components of the core cooling and containment spray systems will be exposed to high temperature borated water. Prolonged exposure to the core cooling water combined with stresses imposed on the components can cause SCC. The SCC is a function of stress, oxygen and chloride concentrations, pH, temperature, and alloy composition of the components. High temperatures and low pH, which would be present after a LOCA, tend to promote SCC. This can lead to the failure of necessary safety systems or components.

Adjusting the pH of the recirculation solution to levels above 7.0 prevents a significant fraction of the dissolved iodine from converting to a volatile form. The higher pH thus decreases the level of airborne iodine in containment and reduces the radiological consequences from containment atmosphere leakage following a LOCA. Maintaining the solution pH above 7.0 also reduces the occurrence of SCC of austenitic stainless steel components in containment. Reducing SCC reduces the probability of failure of components.

BASES

BACKGROUND (continued)

TSP is employed as a passive form of pH control for post LOCA containment spray and core cooling water. Baskets of TSP are placed on the floor or in the sump of the containment building to dissolve from released reactor coolant water and containment sprays after a LOCA. Recirculation of the water for core cooling and containment sprays then provides mixing to achieve a uniform solution pH. The hydrated form (45-57% moisture) of TSP is used because of the high humidity in the containment building during normal operation. Since the TSP is hydrated, it is less likely to absorb large amounts of water from the humid atmosphere and will undergo less physical and chemical change than the anhydrous form of TSP.

APPLICABLE SAFETY ANALYSES

The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being ≥ 7.0. The radionuclide releases from the containment atmosphere and the consequences of a LOCA would be increased if the pH of the recirculation water were not adjusted to 7.0 or above.

TSP satisfies Criterion 3 of the 10 CFR 50.36(c)(2)(ii).

LCO

The TSP is required to adjust the pH of the recirculation water to > 7.0 after a LOCA. A pH > 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH > 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

The required amount of TSP is based upon the extreme cases of water volume and pH possible in the containment sump after a large break LOCA. The minimum required volume is the volume of TSP that will achieve a sump solution pH of ≥ 7.0 when taking into consideration the maximum possible sump water volume and the minimum possible pH. The amount of TSP needed in the containment building is based on the mass of TSP required to achieve the desired pH. However, a required volume is specified, rather than mass, since it is not feasible to weigh the entire amount of TSP in containment. The minimum required volume is based on the manufactured density of TSP. Since TSP can have a tendency to agglomerate from high humidity in the containment building, the density may increase and the volume decrease during normal plant operation. Due to possible agglomeration and increase in density, estimating the minimum volume of TSP in containment is conservative with respect to achieving a minimum required pH.

BASES

APPLICABILITY

with pressurizer pressure ≥ 1750 psia

In MODES 1, 2, and 3, the RCS is at elevated temperature and pressure, providing an energy potential for a LOCA. The potential for a LOCA results in a need for the ability to control the pH of the recirculated coolant. 1

3 with pressurizer pressure < 1750 psia,

In MODES 4, 5, and 6, the potential for a LOCA is reduced or nonexistent, and TSP is not required. 1

ACTIONS

A.1

If it is discovered that the TSP in the containment building sump is not within limits, action must be taken to restore the TSP to within limits. During plant operation the containment sump is not accessible and corrections may not be possible.

The Completion Time of 72 hours is allowed for restoring the TSP within limits, where possible~~, because 72 hours is the same time allowed for restoration of other ECCS components~~. 1

B.1 and B.2

If the TSP cannot be restored within limits within the Completion Time of Required Action A.1, the plant must be brought to a MODE in which the LCO does not apply. The specified Completion Times for reaching MODE~~S~~ 3 and ~~4 are~~ those used throughout the Technical Specifications; they were chosen to allow reaching the specified conditions from full power in an orderly manner and without challenging plant systems. 3

and reducing pressurizer pressure below 1750 psia correspond to

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Periodic determination of the volume of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. A verification is required to determine visually that a minimum of ~~[291]~~ cubic feet is contained in the TSP baskets. This requirement ensures that there is an adequate volume of TSP to adjust the pH of the post LOCA sump solution to a value ≥ 7.0. 2

173

~~[The periodic verification is required every 18 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 18 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.~~ 2

and takes into account that the condition which caused the TSP not within limits would most likely allow the ECCS and Containment Spray System to continue to provide some capability for pH adjustment and iodine removal. The 72 hour Completion Time also considers that the Containment Spray System would continue to be available to remove some iodine from the containment atmosphere in the event of a DBA and the low probability of the worst case DBA occurring during this period.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

SR 3.5.5.2

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of ~~[~~] ~~]~~ grams of TSP from one of the baskets in containment is submerged in ~~1.0 gal ± 0.05~~ gal of water at a boron concentration ~~of~~ [ ] ppm and at the standard temperature of ~~25°C ± 5°C~~. Without agitation, the solution pH should be raised to ≥ 7 within 4 hours. ~~The representative sample weight is based on the minimum required TSP weight of [ ] kilograms, which at manufactured density corresponds to the minimum volume of [ ] cubic ft, and maximum possible post LOCA sump volume of [ ] gallons, normalized to buffer a 1.0 gal sample.~~ The boron concentration of the test water is representative of the ~~maximum possible~~ boron concentration corresponding to the ~~maximum possible~~ post LOCA sump volume. Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. The test time ~~of 4 hours~~ is necessary to allow time for the dissolved TSP to naturally diffuse through the sample solution. In the post LOCA containment sump, rapid mixing would occur, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH ≥ 7.0 by the onset of recirculation after a LOCA.

70.5 ± 0.5
10.0 ± 0.1
between 1900 and 2200
120 ± 10°F
2
1
1
1

~~[The periodic verification is required every 18 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 18 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.~~

~~OR~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

REFERENCES None.

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.5, BASES, TRISODIUM PHOSPHATE (TSP) – UNIT 2 ONLY

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The end state is changed from MODE 4 to MODE 3 with pressurizer pressure < 1750 psia consistent with the CTS Applicability. The ISTS Bases is changed consistent with the change to the ISTS

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.5.5, TRISODIUM PHOSPHATE (TSP) – UNIT 2 ONLY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

ISTS Not Adopted

R1

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

R1

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Trisodium Phosphate (TSP)

LCO 3.5.5 The TSP baskets shall contain ≥ [291] ft^3 of active TSP.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TSP not within limits.	A.1 Restore TSP to within limits.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours [12] hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.5.1 Verify the TSP baskets contain ≥ [291] ft^3 of trisodium phosphate.	[[18] months OR In accordance with the Surveillance Frequency Control Program]

SURVEILLANCE REQUIREMENTS (continued) R1

SURVEILLANCE		FREQUENCY
SR 3.5.5.2	Verify that a sample from the TSP baskets provides adequate pH adjustment of RWT water.	[[18] months OR In accordance with the Surveillance Frequency Control Program

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.5, TRISODIUM PHOSPHATE (TSP) – UNIT 1 ONLY

R1

1. ISTS 3.5.5, “TRISODIUM PHOSPHATE (TSP),” is not included in the St. Lucie Plant Unit 1 ITS because TSP is not included in the PSL Unit 1 design and licensing basis for removal of iodine fission product inventory and control of pH in the containment sump during accident conditions. PSL Unit 1 utilizes the Spray Additive System to remove iodine fission product inventory and control pH in the containment sump during accident conditions.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

1

R1

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Trisodium Phosphate (TSP)

BASES

BACKGROUND

Trisodium phosphate (TSP) is placed on the floor or in the sump of the containment building to ensure that iodine, which may be dissolved in the recirculated reactor cooling water following a loss of coolant accident (LOCA), remains in solution. TSP also helps inhibit stress corrosion cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident.

Fuel that is damaged during a LOCA will release iodine in several chemical forms to the reactor coolant and to the containment atmosphere. A portion of the iodine in the containment atmosphere is washed to the sump by containment sprays. The emergency core cooling water is borated for reactivity control. This borated water causes the sump solution to be acidic. In a low pH (acidic) solution, dissolved iodine will be converted to a volatile form. The volatile iodine will evolve out of solution into the containment atmosphere, significantly increasing the levels of airborne iodine. The increased levels of airborne iodine in containment contribute to the radiological releases and increase the consequences from the accident due to containment atmosphere leakage.

After a LOCA, the components of the core cooling and containment spray systems will be exposed to high temperature borated water. Prolonged exposure to the core cooling water combined with stresses imposed on the components can cause SCC. The SCC is a function of stress, oxygen and chloride concentrations, pH, temperature, and alloy composition of the components. High temperatures and low pH, which would be present after a LOCA, tend to promote SCC. This can lead to the failure of necessary safety systems or components.

Adjusting the pH of the recirculation solution to levels above 7.0 prevents a significant fraction of the dissolved iodine from converting to a volatile form. The higher pH thus decreases the level of airborne iodine in containment and reduces the radiological consequences from containment atmosphere leakage following a LOCA. Maintaining the solution pH above 7.0 also reduces the occurrence of SCC of austenitic stainless steel components in containment. Reducing SCC reduces the probability of failure of components.

1

R1

BASES

BACKGROUND (continued)

TSP is employed as a passive form of pH control for post LOCA containment spray and core cooling water. Baskets of TSP are placed on the floor or in the sump of the containment building to dissolve from released reactor coolant water and containment sprays after a LOCA. Recirculation of the water for core cooling and containment sprays then provides mixing to achieve a uniform solution pH. The hydrated form (45-57% moisture) of TSP is used because of the high humidity in the containment building during normal operation. Since the TSP is hydrated, it is less likely to absorb large amounts of water from the humid atmosphere and will undergo less physical and chemical change than the anhydrous form of TSP.

APPLICABLE SAFETY ANALYSES

The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being ≥ 7.0. The radionuclide releases from the containment atmosphere and the consequences of a LOCA would be increased if the pH of the recirculation water were not adjusted to 7.0 or above.

TSP satisfies Criterion 3 of the 10 CFR 50.36(c)(2)(ii).

LCO

The TSP is required to adjust the pH of the recirculation water to > 7.0 after a LOCA. A pH > 7.0 is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH > 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

The required amount of TSP is based upon the extreme cases of water volume and pH possible in the containment sump after a large break LOCA. The minimum required volume is the volume of TSP that will achieve a sump solution pH of ≥ 7.0 when taking into consideration the maximum possible sump water volume and the minimum possible pH. The amount of TSP needed in the containment building is based on the mass of TSP required to achieve the desired pH. However, a required volume is specified, rather than mass, since it is not feasible to weigh the entire amount of TSP in containment. The minimum required volume is based on the manufactured density of TSP. Since TSP can have a tendency to agglomerate from high humidity in the containment building, the density may increase and the volume decrease during normal plant operation. Due to possible agglomeration and increase in density, estimating the minimum volume of TSP in containment is conservative with respect to achieving a minimum required pH.

1

R1

BASES

APPLICABILITY

In MODES 1, 2, and 3, the RCS is at elevated temperature and pressure, providing an energy potential for a LOCA. The potential for a LOCA results in a need for the ability to control the pH of the recirculated coolant.

In MODES 4, 5, and 6, the potential for a LOCA is reduced or nonexistent, and TSP is not required.

ACTIONS

A.1

If it is discovered that the TSP in the containment building sump is not within limits, action must be taken to restore the TSP to within limits. During plant operation the containment sump is not accessible and corrections may not be possible.

The Completion Time of 72 hours is allowed for restoring the TSP within limits, where possible, because 72 hours is the same time allowed for restoration of other ECCS components.

B.1 and B.2

If the TSP cannot be restored within limits within the Completion Time of Required Action A.1, the plant must be brought to a MODE in which the LCO does not apply. The specified Completion Times for reaching MODES 3 and 4 are those used throughout the Technical Specifications; they were chosen to allow reaching the specified conditions from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Periodic determination of the volume of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. A verification is required to determine visually that a minimum of [291] cubic feet is contained in the TSP baskets. This requirement ensures that there is an adequate volume of TSP to adjust the pH of the post LOCA sump solution to a value ≥ 7.0.

[The periodic verification is required every 18 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 18 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.

R1

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

SR 3.5.5.2

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of [ ] grams of TSP from one of the baskets in containment is submerged in 1.0 gal ± 0.05 gal of water at a boron concentration of [ ] ppm and at the standard temperature of 25°C ± 5°C. Without agitation, the solution pH should be raised to ≥ 7 within 4 hours. The representative sample weight is based on the minimum required TSP weight of [ ] kilograms, which at manufactured density corresponds to the minimum volume of [ ] cubic ft, and maximum possible post LOCA sump volume of [ ] gallons, normalized to buffer a 1.0 gal sample. The boron concentration of the test water is representative of the maximum possible boron concentration corresponding to the maximum possible post LOCA sump volume. Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved TSP to naturally diffuse through the sample solution. In the post LOCA containment sump, rapid mixing would occur, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH ≥ 7.0 by the onset of recirculation after a LOCA.

[The periodic verification is required every 18 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 18 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.

OR

1

R1

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------REVIEWER'S NOTE----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

REFERENCES None.

JUSTIFICATION FOR DEVIATIONS
ITS 3.5.5 BASES, TRISODIUM PHOSPHATE (TSP) – UNIT 1 ONLY

R1

1. ISTS 3.5.5, “TRISODIUM PHOSPHATE (TSP),” is not included in the St. Lucie Plant Unit 1 ITS because TSP is not included in the PSL Unit 1 design and licensing basis for removal of iodine fission product inventory and control of pH in the containment sump during accident conditions. PSL Unit 1 utilizes the Spray Additive System to remove iodine fission product inventory and control pH in the containment sump during accident conditions.

ENCLOSURE 2

VOLUME 11

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.6
CONTAINMENT SYSTEMS

Revision 0

LIST OF ATTACHMENTS

1. ITS 3.6.1, Containment
2. ITS 3.6.2, Containment Air Locks
3. ITS 3.6.3, Containment Isolation Valves
4. ITS 3.6.4, Containment Pressure
5. ITS 3.6.5, Containment Air Temperature
6. ITS 3.6.6, Containment Spray and Cooling Systems
7. ITS 3.6.7, Shield Building
8. ITS 3.6.8, Vacuum Relief Valves
9. ITS 3.6.9, Shield Building Ventilation System (SBVS)
10. ITS 3.6.10, Spray Additive System – Unit 1 Only
11. ISTS Not Adopted

ATTACHMENT 1

ITS 3.6.1, CONTAINMENT

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT ~~VESSEL~~

A02

CONTAINMENT ~~VESSEL INTEGRITY~~

LIMITING CONDITION FOR OPERATION

LCO 3.6.1 3.6.1.1 ~~CONTAINMENT VESSEL INTEGRITY~~ shall be ~~maintained~~.

Containment

OPERABLE

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

A02

ACTION:

Containment inoperable.

containment to OPERABLE status.

ACTION A ~~Without CONTAINMENT VESSEL INTEGRITY,~~ restore ~~CONTAINMENT VESSEL INTEGRITY~~ within one hour or be in ~~at least HOT STANDBY~~ within the next 6 hours and in ~~COLD~~

ACTION B ~~SHUTDOWN~~ within the following 30 hours.

MODE 3

MODE 5

A01

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1 4.6.1.1 ~~CONTAINMENT VESSEL INTEGRITY shall be demonstrated:~~

Insert 1

A02

A04

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. All containment vessel penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open on an intermittent basis under administrative control, and

SR 3.6.3.2
SR 3.6.3.3

See ITS 3.6.3

~~2. All containment vessel equipment hatches are closed and sealed.~~

LA02

b. By verifying that each containment vessel air lock is OPERABLE per Specification 3.6.1.3

See ITS 3.6.2

* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

See ITS 3.6.3

INSERT 1 A02 A04

Perform required visual examinations and leakage rate testing except for containment air lock and secondary containment bypass leakage testing, in accordance with the Containment Leakage Rate Testing Program.

CONTAINMENT SYSTEMS

CONTAINMENT ~~LEAKAGE~~

LIMITING CONDITION FOR OPERATION

~~3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.~~

APPLICABILITY: ~~MODES 1, 2, 3 and 4.~~

A02

ACTION:

~~With the containment leakage rate exceeding the acceptance criteria of the Containment Leakage Rate Testing Program, within 1 hour initiate action to be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the overall leakage rate to less than that specified by the Containment Leakage Rate Testing Program prior to increasing the Reactor Coolant System temperature above 200°F.~~

A05 R1

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

4.6.1.2 The containment leakage rates shall be demonstrated at the required test schedule and shall be determined in conformance with the criteria specified in the Containment Leakage Rate Testing Program.

except for containment air lock and secondary containment bypass leakage testing

A04

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

~~Pages 3/4 6-4 through 3/4 6-9 have been DELETED~~.

~~Page 3/4 6-10 is the next valid page.~~

CONTAINMENT SYSTEMS

CONTAINMENT ~~VESSEL STRUCTURAL INTEGRITY~~

LIMITING CONDITION FOR OPERATION

~~3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.~~

A02

APPLICABILITY: ~~MODES 1, 2, 3 and 4.~~

ACTION:

~~With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.~~

A03

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

4.6.1.6 The structural integrity of the containment vessel shall be determined, in accordance with the C containment Leakage Rate Testing Program, by a visual inspection ~~of the accessible interior and exterior surfaces of the vessel and verifying no apparent changes in appearance of the surfaces or other abnormal degradation~~.

LA01

DEFINITIONS

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

See ITS Chapter 1.0

~~CONTAINMENT VESSEL INTEGRITY~~

~~1.7 CONTAINMENT VESSEL INTEGRITY shall exist when:~~

~~a. All containment vessel penetrations required to be closed during accident conditions are either:~~

~~1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or~~

~~2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed position except for valves that are open on an intermittent basis under administrative control.~~

See ITS Chapter 3.6.3

~~b. All containment vessel equipment hatches are closed and sealed,~~

LA02

~~c. Each containment vessel air lock is in compliance with the requirements of Specification 3.6.1.3,~~

See ITS Chapter 3.6.2

SR 3.6.1.1

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

~~e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.~~

LA02

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be the seal water flow supplied from the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include shared (4 fingered) control element assemblies (CEAs) withdrawn into the upper guide structure (UGS) or evolutions performed with the UGS in place such as CEA latching/unlatching or verification of latching/ unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

See ITS Chapter 1.0

CORE OPERATING LIMITS REPORT (COLR)

1.9a THE COLR is the unit-specific document that provides cycle specific parameter limits for the current operating reload cycle. These cycle-specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these limits is addressed in individual Specifications.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 ~~PRIMARY~~ CONTAINMENT

CONTAINMENT ~~INTEGRITY~~

LIMITING CONDITION FOR OPERATION

A02

LCO 3.6.1 3.6.1.1 ~~Primary CONTAINMENT INTEGRITY~~ shall be ~~maintained~~.

Containment

OPERABLE

Applicability **APPLICABILITY:** MODES 1~~*~~, 2~~*~~, 3, and 4.

LA03

ACTION:

Containment inoperable.

containment to OPERABLE status.

A02

ACTION A ~~Without primary CONTAINMENT INTEGRITY,~~ restore ~~CONTAINMENT INTEGRITY~~ within 1 hour or be in ~~at least HOT STANDBY~~ within the next 6 hours and in ~~COLD SHUTDOWN~~

ACTION B within the following 30 hours.

MODE 3

MODE 5

A01

SURVEILLANCE REQUIREMENTS

A02

A04

SR 3.6.1.1 4.6.1.1 ~~Primary CONTAINMENT INTEGRITY shall be demonstrated:~~

Insert 1

SR 3.6.3.2
SR 3.6.3.3

a. In accordance with the Surveillance Frequency Control Program by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open on an intermittent basis under administrative control.

See ITS 3.6.3

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

See ITS 3.6.2

~~*~~ ~~In MODES 1 and 2, the RCB polar crane shall be rendered inoperable by locking the power supply breaker open.~~

LA03

** Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

See ITS 3.6.3

INSERT 1 A02 A04

Perform required visual examinations and leakage rate testing except for containment air lock and secondary containment bypass leakage testing, in accordance with the Containment Leakage Rate Testing Program.

CONTAINMENT SYSTEMS

CONTAINMENT ~~LEAKAGE~~

LIMITING CONDITION FOR OPERATION

~~3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.~~

APPLICABILITY: ~~MODES 1, 2, 3, and 4.~~

A02

ACTION:

~~With the containment leakage rate exceeding the acceptance criteria of the Containment Leakage Rate Testing Program, within 1 hour initiate action to be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the overall leakage rate to less than that specified by the Containment Leakage Rate Testing Program, prior to increasing the Reactor Coolant System temperature above 200°F.~~

R1

A05

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

4.6.1.2 The containment leakage rates shall be demonstrated at the required test schedule and shall be determined in conformance with the criteria specified in the Containment Leakage Rate Testing Program.

except for containment air lock and secondary containment bypass leakage testing

A04

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

~~Pages 3/4 6-4 through 3/4 6-8 have been DELETED.~~

~~Page 3/4 6-9 is the next valid page.~~

CONTAINMENT SYSTEMS

CONTAINMENT ~~VESSEL STRUCTURAL INTEGRITY~~

LIMITING CONDITION FOR OPERATION

~~3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Surveillance Requirement 4.6.1.6.~~

A02

APPLICABILITY: ~~MODES 1, 2, 3 and 4.~~

ACTION:

~~With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.~~

A03

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1 4.6.1.6 The structural integrity of the containment ~~-~~vessel shall be determined in accordance with the Containment Leak Rate Testing Program by a visual inspection~~ of the exposed accessible interior and exterior surfaces of the vessel and verifying no apparent changes in appearance of the surfaces or other abnormal degradation~~.

LA01

DEFINITIONS

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

~~### CONTAINMENT VESSEL INTEGRITY~~

~~1.7 CONTAINMENT VESSEL INTEGRITY shall exist when:~~

See ITS Chapter 3.6.3

See ITS Chapter 1.0

~~a. All containment vessel penetrations required to be closed during accident conditions are either:~~

~~1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or~~

~~2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open on an intermittent basis under administrative control.~~

~~b. All containment vessel equipment hatches are closed and sealed,~~

LA02

~~c. Each containment vessel air lock is in compliance with the requirements of Specification 3.6.1.3,~~

See ITS Chapter 3.6.2

SR 3.6.1.1

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

~~e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.~~

LA02

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be the seal water flow supplied from the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include evolutions performed with the upper guide structure (UGS) in place such as control element assembly (CEA) latching/unlatching or verification of latching/unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

See ITS Chapter 1.0

1.9a THE COLR is the unit-specific document that provides cycle specific parameter limits for the current operating reload cycle. These cycle-specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these limits is addressed in individual Specifications.

DISCUSSION OF CHANGES
ITS 3.6.1, CONTAINMENT

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3/4.6.1 requires CONTAINMENT VESSEL INTEGRITY. CTS 3.6.1.1 states "CONTAINMENT VESSEL INTEGRITY shall be maintained." Unit 1 CTS 3.6.1.1 ACTION requires, in part, “without CONTAINMENT VESSEL INTEGRITY to restore CONTAINMENT VESSEL INTEGRITY within one hour."

Unit 2 CTS 3/4.6.1 requires PRIMARY CONTAINMENT INTEGRITY. Unit 2 CTS 3.6.1.1 states "Primary CONTAINMENT INTEGRITY shall be maintained." Unit 2 CTS 3.6.1.1 ACTION requires, in part, without primary CONTAINMENT INTEGRITY to restore CONTAINMENT INTEGRITY within 1 hour."

Unit 1 and 2 CTS 4.6.1.2 require containment leakage rates to be demonstrated at the required test schedule and in conformance with Containment Leakage Rate Testing Program criteria. Unit 1 and 2 CTS 3.6.1.2 require containment leakage rates be limited in accordance with the Containment Leakage Rate Testing Program.

Unit 1 CTS 4.6.1.6 requires CONTAINMENT VESSEL STRUCTURAL INTEGRITY to be demonstrated by performance of a visual inspection in accordance with the Containment Leak Rate Testing Program. Unit 2 CTS 4.6.1.6 requires the structural integrity of the containment vessel be determined in accordance with the Containment Leak Rate Testing Program by a visual inspection. CTS 3.6.1.6 requires the structural integrity of the containment to be maintained within specified parameters.

ITS 3.6.1 is the containment specification. ITS LCO 3.6.1 requires the containment to be OPERABLE. ITS 3.6.1 ACTION A requires when containment is inoperable to restore the containment to OPERABLE status within 1 hour. ITS SR 3.6.1.1 requires performance of a visual examination and leak rate testing in accordance with the Containment Leak Rate Testing Program, with an exception for the containment air lock and secondary containment bypass leakage testing.

This changes the Unit 1 and 2 CTS by replacing the specific CONTAINMENT VESSEL INTEGRITY definition and all references to it with the requirement for Containment OPERABILITY. Additionally, it changes the Unit 1 and Unit 2 CTS by combining CTS 3.6.1.1, CTS 3.6.1.2, and CTS 3.6.1.6 into one specification.

The purpose of CTS 3.6.1.1, CTS 3.6.1.2 and CTS 3.6.1.6 is to provide requirements pertaining to containment OPERABILITY. This portion of the

change (combining CTS 3.6.1.1, CTS 3.6.1.2 and 3.6.1.6) is acceptable because moving these requirements into one LCO, ITS LCO 3.6.1, centralizes the requirements. The Unit 1 CTS 3/4.6.1 references to CONTAINMENT VESSEL INTEGRITY and the Unit 2 CTS 3/4.6.1 reference to primary CONTAINMENT INTEGRITY have been deleted because the CTS definition of CONTAINMENT VESSEL INTEGRITY in CTS 1.7 is incorporated into ITS 3.6.1, 3.6.2 and 3.6.3 and is no longer maintained as a separate definition in the ITS. ITS 3.6.1 requires that the containment shall be OPERABLE. The definition of OPERABLE and the subsequent ITS 3.6.1 LCO, ACTIONS, and Surveillance Requirements are sufficient to encompass the applicable requirements of the CTS definition. This change removes any confusion that may exist between the definition and the specific requirements of the LCO and is a presentation preference consistent with NUREG-1432, Rev. 5.0. Because the aspects of the CONTAINMENT VESSEL INTEGRITY definition requirements, along with the remainder of the LCOs in the Containment Systems Primary Containment section (i.e., air locks and containment isolation valves), are maintained in subsequent Specifications of ITS, this change is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 3.6.1.6, ACTION, states, "With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F." CTS 3.6.1.6 ACTION does not state what action to take if the structural integrity limits are not met while in MODE 1, 2, 3, or 4. Thus, entry into CTS 3.0.3 is required if CTS 3.6.1.6 is not met while in MODE 1, 2, 3, or 4. CTS 3.0.3 allows 1 hour to prepare for a shutdown and requires the unit to be in HOT STANDBY (ITS MODE 3) within the next 6 hours, HOT SHUTDOWN (ITS MODE 4) within the following 6 hours, and Cold Shutdown (similar to ITS MODE 5) within the subsequent 24 hours (37 hours total). ITS 3.6.1 ACTION A requires that if the containment is inoperable, it must be restored to OPERABLE status within 1 hour. ITS 3.6.1 ACTION B requires that if the Required Action and associated Completion Time are not met (i.e., the containment is not restored to OPERABLE status in 1 hour), the unit must be in MODE 3 within 6 hours and MODE 5 within 36 hours (37 hours total). This changes CTS by stating the ACTIONS rather than deferring to CTS 3.0.3. In addition, it deletes the CTS Actions to restore the limits prior to increasing the Reactor Coolant System temperature above 200°F.

The purpose of CTS 3.0.3 is to place the unit outside the MODE of Applicability within a reasonable amount of time in a controlled manner. CTS 3.6.1.6 is silent on these actions, deferring to CTS 3.0.3 for the actions to accomplish this. This change is acceptable because the ACTIONS specified in ITS 3.6.1 adopt ISTS structure for placing the unit outside the MODE of Applicability while changing the time specified to enter MODE 3 and MODE 5 but still within the plants ability to safely shutdown. In addition, deletion of the current Actions of CTS 3.6.1.6 is acceptable, because CTS 4.0.4 (ITS SR 3.0.4) already precludes entering the MODE of Applicability when the LCO is not met. Therefore, it is not necessary to include these requirements as specific actions in ITS 3.6.1. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 4.6.1.2 requires containment leakage rates be demonstrated in conformance with the criteria specified in the Containment Leakage Rate Testing Program. ITS SR 3.6.1.1 requires the same test but adds an exception for containment air lock testing and secondary containment bypass leakage. This changes the Unit 1 and Unit 2 CTS by excluding containment air lock and secondary containment bypass leakage testing.

This is acceptable because ITS SR 3.6.2.1 requires performance of air lock leakage testing, Unit 1 ITS SR 3.6.3.7 and Unit 2 ITS SR 3.6.3.8 require secondary containment bypass leakage testing. Individual leakage rates specified for the containment air lock (CTS 4.6.1.1.b and 4.6.1.3.a) and secondary bypass leakage (CTS 6.8.4.h) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 La. This change is designated as administrative because it does not result in technical changes to the CTS.

A05 CTS 3.6.1.2 Action, in part, requires restoring the overall leakage rate to less than that specified by the Containment Leakage Rate Testing Program prior to increasing the Reactor Coolant System temperature above 200°F (i.e., MODE 4). Although not explicitly stated, if overall leakage rate is not within the limits of the Containment Leakage Rate Testing Program prior to increasing the RCS temperature above 200°F, the LCO is not met and CTS 3.0.4 (ITS LCO 3.0.4) would apply. ITS 3.6.1 ACTIONS do not include an explicit requirement to restoring the overall leakage rate to less than that specified by the Containment Leakage Rate Testing Program prior to increasing the Reactor Coolant System temperature above 200°F because it is inherently required by ITS LCO 3.0.4. This changes the CTS by deleting the Action to restore the LCO prior to increasing RCS temperature above 200°F (i.e., entering MODE 4). R1

This change is acceptable because CTS 3.0.4 (ITS LCO 3.0.4) already precludes entering the MODE of Applicability when the LCO is not met except: a) when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; b) after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; or c) when an allowance is stated in the individual value, parameter, or other Specification. With overall leakage rate not within the limits specified in the Containment Leakage Rate Testing Program, none of the allowances provided in CTS 3.0.4 apply because the Technical Specification actions do not allow operation in MODE 4 for an unlimited period of time, a risk assessment cannot be performed and managed on a parameter outside limits, and there is no specific allowance stated in the Specification. Therefore, it is not necessary to include explicit requirements to restore the overall leakage rate to less than that specified by the Containment Leakage Rate Testing Program prior to increasing the Reactor Coolant System temperature above 200°F. This change is designated as administrative, because it does not result in a technical change to the CTS.

DISCUSSION OF CHANGES
ITS 3.6.1, CONTAINMENT

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* Unit 1 CTS 4.6.1.6 states , "The structural integrity of the containment vessel shall be determined, in accordance with the containment Leakage Rate Testing Program, by a visual inspection of the accessible interior and exterior surfaces of the vessel and verifying no apparent changes in appearance of the surfaces or other abnormal degradation." Unit 2 CTS 4.6.1.6 states , "The structural integrity of the containment vessel shall be determined in accordance with the Containment Leak Rate Testing Program by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel and verifying no apparent changes in appearance of the surfaces or other abnormal degradation."

ITS SR 3.6.1.1 requires the same visual examination in accordance with the Containment Leakage Rate Testing Program but does not include the details of when the testing is performed, what is verified and the acceptance criteria. This changes the CTS by moving the details of the visual examination to the Containment Leakage Rate Testing Program.

The removal of these details associated with the visual examination of the containment vessel is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. ITS SR 3.6.1.1 retains the requirement to perform a visual examination in accordance with the Containment Leakage Rate Testing Program. This change is acceptable because these types of procedural details will be adequately controlled in the Containment Leakage Rate Testing Program requirements in ITS Chapter 5. This change is designated as a less restrictive removal of detail change because details associated with performance of the containment vessel visual examination are being removed from the Technical Specifications.

LA02 *(Type 2 – Removing Descriptions of System Operation)* CTS 1.7 states, in part, "CONTAINMENT VESSEL INTEGRITY shall exist when: a. All penetrations required to be closed during accident conditions are either: 1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or 2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control. b. All containment vessel equipment hatches are closed and sealed, c. Each containment vessel air lock is in compliance with the requirements of Specification 3.6.1.3, and e. The sealing mechanism associated

with each penetration (e.g., welds, bellows, or 0-rings) is OPERABLE." ITS 3.6.1 states "Containment shall be OPERABLE." This changes the CTS by moving the reference to penetration and containment vessel equipment hatches and sealing mechanism associated with each penetration to the Bases.

The removal of these details, that are related to system operation, from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for the containment to be OPERABLE and the relocated material describes aspects of OPERABILITY. In addition, the ITS retains the requirement to perform required visual inspections and leakage rate testing in accordance with the Containment Leakage Rate Testing Program in accordance with 10 CFR 50 Appendix J, Part B, that would provide verification that the equipment hatch is closed and sealed and the sealing mechanisms are OPERABLE. Also, this change is acceptable, because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system operation is being removed from the Technical Specifications.

LA03 **Unit 2 only:** *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.6.1.1, by footnote, states the APPLICABILITY to require "In MODES 1 and 2, the RCB polar crane shall be rendered inoperable by locking the power supply breaker open." The ITS 3.6.1 APPLICABILITY is "MODES 1, 2, 3, and 4." This changes the CTS by moving the procedural detail to the UFSAR.

The removal of these details, for performing actions, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for the containment to be OPERABLE in MODES 1, 2, 3, and 4 and the relocated material describes procedural requirements of OPERABILITY. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment ~~(Atmospheric and Dual)~~ 1

3.6.1.1 3.6.1.2 3.6.1.6 LCO 3.6.1 Containment shall be OPERABLE.

Applicability (3.6.1.1 3.6.1.2 3.6.1.6) APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.6.1.1 ACTION 3.6.1.2 ACTION 3.6.1.6 ACTION A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
3.6.1.1 ACTION 3.6.1.2 ACTION 3.6.1.6 ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.6.1.2 4.6.1.6 SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, and secondary containment bypass leakage in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program 2
~~SR 3.6.1.2~~	~~[Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.~~	~~In accordance with the Containment Tendon Surveillance Program]~~ 3

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment ~~(Atmospheric and Dual)~~ 1

3.6.1.1 3.6.1.2 3.6.1.6 LCO 3.6.1 Containment shall be OPERABLE.

Applicability (3.6.1.1 3.6.1.2 3.6.1.6) APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.6.1.1 ACTION 3.6.1.2 ACTION 3.6.1.6 ACTION A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
3.6.1.1 ACTION 3.6.1.2 ACTION 3.6.1.6 ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.6.1.2 4.6.1.6 SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, and secondary containment bypass leakage in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program 2
~~SR 3.6.1.2~~	~~[Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.~~	~~In accordance with the Containment Tendon Surveillance Program]~~ 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.1, CONTAINMENT

1. The type of Containment (Atmospheric and Dual) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ITS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants or may not be applicable to all Combustion Engineering plants. Since St. Lucie Plant (PSL) Units 1 and 2 do not have containment tendons, ISTS SR 3.6.1.2 is not included in PSL Units 1 and 2 ITS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1~~B~~ Containment ~~(Dual)~~ 1

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom, completely enclosed by a reinforced concrete shield building. A 4 ft wide annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to permit inservice inspection and collection of containment outleakage. Dual containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment OPERABILITY. The shield building provides biological shielding and allows controlled release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and the Nuclear Steam Supply System.

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. Loss of containment OPERABILITY could cause site boundary doses, in the event of a Design Basis Accident (DBA), to exceed values given in the licensing basis. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option ~~[A][~~B~~]~~ (Ref. 1), as modified by approved exemptions. 3

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

BASES

BACKGROUND (continued)

a. All penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE automatic containment isolation system or

2. Closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves,"

b. Each air lock is OPERABLE except as provided in LCO 3.6.2, "Containment Air Locks,"

c. All equipment hatches are closed, and

Insert 1

~~[~~d. ~~The pressurized sealing mechanism associated with a penetration, except as provided in LCO 3.6.[ ], is OPERABLE.]~~ 3

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a release of radioactive material within containment are a LOCA, a main steam line break, and a control element assembly ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[~~0.50~~]~~% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option ~~[A][~~B~~]~~ (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of ~~[42.3]~~ psig, which results from the limiting design basis LOCA (Ref. ~~2~~). 3

42.77

3 2

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

St. Lucie – Unit 1

3 **INSERT 1**

The sealing mechanism associated with each containment penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

BASES

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) ~~[~~, purge valves with resilient seals, and secondary bypass leakage (LCO 3.6.3)~~]~~ are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a. 3

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 refueling operations are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event that containment is inoperable, it must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment ~~concrete~~ visual examinations may be performed during ~~either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during~~ a maintenance or refueling outage. ~~The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.~~ 2

Failure to meet air lock and purge valve with resilient seal specific leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and ~~[< 0.75 L_a for Option A] [~~≤ 0.75 L_a ~~for Option B]~~ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis. 3

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

~~REVIEWER'S NOTE~~
~~Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.~~ 4

~~[SR 3.6.1.2~~

~~For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are in accordance with the ASME Code, Section XI, Subsection IWL (Ref. 4), and applicable addenda as required by 10 CFR 50.55a.]~~ 5

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option ~~[A]~~[B~~]~~.
2. FSAR, Section ~~[~~ ~~]~~. (U, 15)
3. FSAR, Section ~~[~~ ~~]~~. (U, 6.2)

~~4. ASME Code, Section XI, Subsection IWL.~~ (5)

(3, 6)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1~~B~~ Containment ~~(Dual)~~ 1

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom, completely enclosed by a reinforced concrete shield building. A 4 ft wide annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to permit inservice inspection and collection of containment outleakage. Dual containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment OPERABILITY. The shield building provides biological shielding and allows controlled release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and the Nuclear Steam Supply System.

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. Loss of containment OPERABILITY could cause site boundary doses, in the event of a Design Basis Accident (DBA), to exceed values given in the licensing basis. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option ~~[A][~~B~~]~~ (Ref. 1), as modified by approved exemptions. 3

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

BASES

BACKGROUND (continued)

a. All penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE automatic containment isolation system or

2. Closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves,"

b. Each air lock is OPERABLE except as provided in LCO 3.6.2, "Containment Air Locks,"

c. All equipment hatches are closed, and

Insert 1

~~[~~d. ~~The pressurized sealing mechanism associated with a penetration, except as provided in LCO 3.6.[ ], is OPERABLE.]~~ 3

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a release of radioactive material within containment are a LOCA, a main steam line break, and a control element assembly ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[~~0.50~~]~~% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option ~~[A][~~B~~]~~ (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of ~~[42.3]~~ psig, which results from the limiting design basis LOCA (Ref. ~~2~~). 3

43.43

3 2

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

St. Lucie – Unit 2

3 **INSERT 1**

The sealing mechanism associated with each containment penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

BASES

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) ~~[~~, purge valves with resilient seals, and secondary bypass leakage (LCO 3.6.3)~~]~~ are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a. 3

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 refueling operations are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event that containment is inoperable, it must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment ~~concrete~~ visual examinations may be performed during ~~either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during~~ a maintenance or refueling outage. ~~The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.~~ 2

Failure to meet air lock and purge valve with resilient seal specific leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and ~~[< 0.75 L_a for Option A] [~~≤ 0.75 L_a ~~for Option B]~~ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis. 3

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

~~REVIEWER'S NOTE~~
~~Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.~~ 4

~~[SR 3.6.1.2~~

~~For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are in accordance with the ASME Code, Section XI, Subsection IWL (Ref. 4), and applicable addenda as required by 10 CFR 50.55a.]~~ 5

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option ~~[A]~~[B~~]~~.
2. FSAR, Section ~~[~~ ~~]~~. (U, 15)
3. FSAR, Section ~~[~~ ~~]~~. (U, 6.2)

~~4. ASME Code, Section XI, Subsection IWL.~~ (5)

(3, 6)

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.1 Bases, CONTAINMENT

1. The heading for ISTS 3.6.1 includes the parenthetical expression (Dual) and the Specification designator “B” that are deleted. This identifying information is not included in the St. Lucie Plant (PSL) ITS. This information is provided in the NUREG to assist in identifying the appropriate Specification to be used as a model for a plant specific ITS conversion but serves no purpose in a plant specific implementation. In addition, the Atmospheric Specification Bases (ISTS B 3.6.1A) is not used and not shown.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is changed to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. This bracketed requirement regarding Containment Tendon Surveillance Program is deleted because it is not applicable to PSL Unit 1 and Unit 2. The PSL containments do not utilize containment tendons.

6. The PSL Safety Analysis Report (SAR) is an updated version of the original Final Safety Analysis Report. Therefore, the proper acronym is UFSAR and is changed to reflect the document.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.6.1, CONTAINMENT

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

ITS 3.6.2, CONTAINMENT AIR LOCKS

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

LCO 3.6.2 3.6.1.3 ~~Each~~ containment air lock~~s~~ shall be OPERABLE ~~with:~~

Two

A02

~~a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and~~

LA01

SR 3.6.2.1 b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTIONS **ACTION:**

Add proposed ACTION Note 1

NOTES

NOTE

ACTION NOTE 1 • ~~If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door.~~ No more than one air lock door shall be open at any time.

L01

A03

ACTION A.1, B.1, C.2

Add proposed ACTION Note 2

ACTION NOTE 3 • Enter ~~the ACTION~~ of LCO 3.6.1~~.2~~, "Containment ~~Leakage~~," when ~~air lock~~ leakage results in exceeding overall containment leakage rate acceptance criteria.

A01

L02

applicable Conditions and Required Actions

Add proposed ACTION A Required Action Note 1

Add proposed ACTION A Required Action Note 2

Add proposed Required Action A.1 Completion Time

ACTION A a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and ~~either restore the inoperable air lock door to OPERABLE status~~ within 24 hours or lock the OPERABLE air lock door closed.

A04

M01

Add proposed Required Action A.3 Note

L02

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be closed at least once per 31 days.

Add proposed ACTION B

L01

ACTION D 3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Add proposed Required Action C.1

M02

ACTION C b. With one or both containment air lock(s) inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed ~~in the affected air lock(s)~~ and restore the inoperable air lock(s) to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program;

ACTION D otherwise be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Add proposed Required Action C.2 Completion Time

M01

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

SURVEILLANCE REQUIREMENTS (continued)

Add proposed SR 3.6.2.1 Notes 1 and 2

A05 | R1

SR 3.6.2.1 a. By verifying leakage rates and air lock door seals in accordance with the Containment Leakage Rate Testing Program; and

SR 3.6.2.2 b. In accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT VESSEL

CONTAINMENT VESSEL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without CONTAINMENT VESSEL INTEGRITY, restore CONTAINMENT VESSEL INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.6.1

SURVEILLANCE REQUIREMENTS

See ITS 3.6.3

4.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be demonstrated:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. All containment vessel penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open on an intermittent basis under administrative control, and

2. All containment vessel equipment hatches are closed and sealed.

SR 3.6.2.1

b. By verifying that each containment vessel air lock is OPERABLE per Specification 3.6.1.3

See ITS 3.6.3

* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

LCO 3.6.2 3.6.1.3 ~~Each~~ containment air lock shall be OPERABLE ~~with:~~

Two

s

A02

a. ~~Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and~~

LA01

SR 3.6.2.1 b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

Applicability **APPLICABILITY:** MODES 1, 2, 3, and 4.

ACTIONS **ACTION:**

Add proposed ACTION Note 1

NOTES

NOTE

ACTION NOTE 1

- ~~If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door.~~ No more than one airlock door shall be open at any time.

L01

A03

ACTION A.1, B.1, C.2

Add proposed ACTION Note 2

ACTION NOTE 3

- Enter ~~the ACTION~~ of LCO 3.6.1~~.2~~, "Containment ~~Leakage~~," when ~~air lock~~ leakage results in exceeding overall containment leakage rate acceptance criteria.

A01

L02

applicable Conditions and Required Actions

Add proposed ACTION A Required Action Note 1

Add proposed ACTION A Required Action Note 2

Add proposed Required Action A.1 Completion Time

ACTION A

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and ~~either restore the inoperable air lock door to OPERABLE status~~ within 24 hours or lock the OPERABLE air lock door closed.

A04

M01

Add proposed Required Action A.3 Note

L02

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

Add proposed ACTION B

L01

ACTION D

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Add proposed Required Action C.1

M02

ACTION C

b. With one or both containment air lock(s) inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed ~~in the affected air lock(s)~~ and restore the inoperable air lock(s) to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION D

Add proposed Required Action C.2 Completion Time

M01

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

Add proposed SR 3.6.2.1 Notes 1 and 2

A05

SR 3.6.2.1 a. By verifying leakage rates and air lock door seals in accordance with the Containment Leakage Rate Testing Program; and

SR 3.6.2.2 b. In accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1*, 2*, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.6.1

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

See ITS 3.6.3

a. In accordance with the Surveillance Frequency Control Program by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open on an intermittent basis under administrative control.

SR 3.6.2.1

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

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* In MODES 1 and 2, the RCB polar crane shall be rendered inoperable by locking the power supply breaker open.

See ITS 3.6.1

** Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

See ITS 3.6.3

DISCUSSION OF CHANGES
ITS 3.6.2, CONTAINMENT AIR LOCKS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.6.1.3 states, in part, "Each containment air lock shall be OPERABLE. ITS LCO 3.6.2 states, "Two containment air locks shall be OPERABLE. This changes the CTS by identifying the total number of airlocks that are required to be OPERABLE.

This change is acceptable because PSL has two containment air locks. Therefore, requiring two containment air locks to be OPERABLE in ITS 3.6.2 is the same as requiring each containment air lock to be OPERABLE in CTS 3.6.1.3. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 3.6.1.3 states, in part, "Each containment air lock shall be OPERABLE." CTS 3.6.1.3 ACTION a states, in part, "With one containment air lock door inoperable" and specifies ACTIONS to be taken. CTS 3.6.1.3 ACTION b states, in part, "With one or both containment air lock(s) inoperable, except as the result of an inoperable air lock door" and specifies ACTIONS to be taken. ITS 3.6.2 ACTIONS Note 2 states "Separate Condition entry is allowed for each airlock." ITS 3.6.2 ACTION A states "One or more containment air locks with one containment air lock door inoperable." ITS 3.6.2 ACTION C states "One or more containment air locks inoperable for reasons other than Condition A or B." This changes the CTS by clarifying the current intent of applying the CTS Actions to each air lock separately.

The purpose of the Unit 1 CTS 3.6.1.3 is to ensure containment air locks meet their requirements for CONTAINMENT VESSEL INTEGRITY and Unit 2 CTS 3.6.1.3 is to ensure containment air locks meet their requirements for primary CONTAINMENT INTEGRITY. One OPERABLE air lock door in each containment air lock provides a pressure boundary and applying the CTS Actions for an inoperable air lock to each of the air locks separately is appropriate. ITS 3.6.2 ACTIONS Note 2 clearly states this. The Required Actions for each Condition provides appropriate compensatory action for each air lock. This change is acceptable because it clarifies existing requirements and better describes how the requirements are currently used. This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 3.6.1.3 ACTION a.1 states in part, to either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed. ITS 3.6.2 ACTION A does not contain the statement to restore the inoperable air lock door to OPERABLE status. This changes CTS by not

including the statement to restore the inoperable air lock door to OPERABLE status.

This change is acceptable because the technical requirements have not changed. Restoration to compliance with the LCO is always an available Required Action and it is the convention in the ITS to not state such "restore" options explicitly unless it is the only action or is required for clarity. This change is designated as administrative because it does not result in technical changes to the CTS.

A05 CTS 4.6.1.3.a requires demonstrating each containment air lock is OPERABLE by verifying leakage rates are in accordance with the Containment Leakage Rate Testing Program. ITS 3.6.2.1 requires the same test, but adds two Notes (SR 3.6.2.1 Note 1) that states in part than an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test and Note 2) requiring that the results be evaluated against acceptance criteria of ITS SR 3.6.1.1. This changes the CTS by clarifying the validity of the previous overall air lock leakage test results and specifically requiring verification of the air lock leakage rates against the Containment leakage rates.

The purpose of leak rate testing requirements for air lock leakage (Type B leakage test) is to verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. Evaluating the containment air lock leakage with the containment leakage rates ensures that the air lock leakage is accounted for in determining the combined Type B and C containment leakage rate. In the CTS, this evaluation is performed as part of the Containment Leakage Rate Testing Program, but is not specifically addressed in CTS 4.6.1.3.a. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.6.1.3 ACTION a requires, in part, maintaining at least the OPERABLE air lock door closed when one containment air lock door is inoperable. CTS 3.6.1.3 ACTION b requires, in part, maintaining at least one air lock door closed when the containment air lock is inoperable. ITS 3.6.2 ACTIONS A and C require similar actions (Required Action A.1 and C.2, respectively), and require verifying the door is closed in the affected air lock within 1 hour. This changes the CTS by adding a new Completion Time.

The purpose of ITS 3.6.2 Required Actions A.1 and C.2 is to verify that the overall leakage rate aspect of containment OPERABILITY is met in the event an airlock is inoperable for a reason other than an interlock mechanism being inoperable. An additional purpose of ITS 3.6.2 Required Actions A.1 and C.2 is to minimize, to the extent possible, the leakage through the inoperable air lock. This change is acceptable because if the inoperability is something that could cause the overall containment leakage rate limits to be exceeded, this should be evaluated immediately, commensurate with the importance of the limits. Furthermore, the one hour Completion Time is commensurate with the Completion Time in ITS 3.6.1 (CTS 3.6.1.1) for restoring containment to

OPERABLE status when the containment is inoperable. This change is considered more restrictive because it provides a new Completion Time.

M02 CTS 3.6.1.3 ACTION b requires maintaining at least one air lock door closed and restoration of an inoperable air lock within 24 hours. ITS 3.6.2 ACTION C requires one additional Required Action. When one or more containment air locks are inoperable for reasons other than Condition A or B, ITS 3.6.2 Required Action C.1 requires initiation of action to evaluate overall containment leakage rate per LCO 3.6.1, immediately. This changes the CTS by adding a new Required Action.

The purpose of ITS 3.6.2 Required Action C.1 is to verify that the overall leakage rate aspect of containment OPERABILITY is met in the event an airlock is inoperable for a reason other than one door or an interlock mechanism being inoperable. An additional purpose of ITS 3.6.2 Required Action C.2 is to minimize, to the extent possible, the leakage through the inoperable air lock. This change is acceptable because if the inoperability is something that could cause the overall containment leakage rate limits to be exceeded, this should be evaluated immediately, commensurate with the importance of the limits. This change is considered more restrictive because it provides a new Required Action.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.6.1.3 states each containment air lock shall be OPERABLE with both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed. ITS 3.6.2 does not contain this level of detail. This changes the CTS by moving details concerning what constitutes an OPERABLE containment air lock to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the information and is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.6.1.3 ACTION b states "With one or both containment air lock(s) inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed in the affected air lock(s) and restore the inoperable air lock(s) to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in at least HOT STANDBY within the next six hours and in COLD SHUTDOWN within the following 6 hours." CTS 3.6.1.3 ACTION Note(s) state in part “If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door.” ITS 3.6.2 ACTION B provides a separate ACTION for inoperability of the air lock with an air lock interlock mechanism inoperable, ITS 3.6.2 ACTION B allows unlimited operation provided an OPERABLE door in the air lock is closed in 1 hour, locked closed in 24 hours, and verification is performed every 31 days that an OPERABLE air lock door in the air lock remains closed. For air lock doors in high radiation areas, this 31 day verification can be performed by administrative means. In addition, containment entry and exit through the air lock is permissible (i.e., the closed and locked door can be opened) under administrative control. Additionally, a new Note which applies to ITS 3.6.2 ACTIONS A, B, and C has been added. This Note, ITS 3.6.2 ACTIONS Note 1, states entry and exit (i.e., the closed and locked OPERABLE air locks can be opened) is permissible to perform repairs on the affected air lock components. This changes the CTS by allowing unlimited operation, with certain restrictions, for air locks that are inoperable due to an inoperable lock mechanism and allows entry and exit to repair an inoperable door.

The purpose of CTS air lock ACTION b is to ensure the containment is not allowed to operate indefinitely in a condition such that it cannot perform its safety function. The changes are acceptable because the proposed ACTION will still ensure the containment safety function is met. Since there are two redundant doors in each air lock, only one OPERABLE air lock door is needed to be maintained closed to ensure the leak tightness requirements are met. The leak tightness of each door is verified, as required by ITS SR 3.6.2.1, in accordance with the Containment Leakage Rate Testing Program. In addition, the interlock mechanism only ensures that both doors in the air lock are not inadvertently opened at the same time. With either an OPERABLE air lock door locked closed, or a dedicated individual ensuring that only one door at a time is opened, the function of the interlock mechanism is being met. The allowances to open the air lock doors to perform repairs or other reasons is acceptable since the time the door is opened is short and the opening is under administrative controls. Also, for the case where the air lock door is opened for reasons other than to effect repairs, the time period (7 days) is short. These changes are designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.6.1.3 ACTION a states "With one containment air lock inoperable, maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed” and “Operation may then continue until performance of the next required overall air

lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days." ITS 3.6.2 ACTION A contains similar requirements, but contains two Required Action Notes stating "Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered" and "Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable." Additionally ITS 3.6.2 Required Action A.3 contains a Note stating "Air lock doors in high radiation areas may be verified locked closed by administrative means." This changes the CTS by ensuring that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable, allowing use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door, and allowing air lock doors in high radiation areas to be verified locked close by administrative means.

The addition of ITS 3.6.2 Condition A, Required Action Note 1 is acceptable since it ensures that only the Required Actions and associated Completion Times of Condition C are performed if both doors in the same air lock are inoperable. This is acceptable since Condition C contains the appropriate remedial actions to take when both doors are inoperable. The addition of ITS 3.6.2 Condition A, Required Action Note 2 is acceptable since it allows entry and exit to perform Technical Specification Surveillance and Required Action as well as other activities on equipment inside containment that are required by Technical Specifications or activities on equipment that support Technical Specification required equipment. This change is acceptable since the time duration is short and the doors are opened under administrative controls. The addition of ITS 3.6.2 Required Action A.3 Note is acceptable since it allows verifying that air lock doors in high radiation areas by administrative means. Additionally, this change is acceptable since access to these areas is typically restricted and the probability of a misalignment of the doors is small. These changes are designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

1

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks ~~(Atmospheric and Dual)~~

1

3

3.6.1.3 LCO 3.6.2 ~~[~~Two~~]~~ containment air lock~~[s]~~ shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

----------NOTES----------

DOC L01 1. Entry and exit is permissible to perform repairs on the affected air lock components.

DOC A03 2. Separate Condition entry is allowed for each air lock.

ACTION NOTE 3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a A. One or more containment air locks with one containment air lock door inoperable.	----------NOTES---------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls ~~[~~if both air locks are inoperable~~]~~. ---------- A.1 Verify the OPERABLE door is closed in the affected air lock. <u>AND</u>	3 DOC L02 DOC M01 1 hour

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2 Lock the OPERABLE door closed in the affected air lock. AND A.3 ----------NOTE---------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- Verify the OPERABLE door is locked closed in the affected air lock.	24 hours Once per 31 days
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	----------NOTES---------- 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. ---------- B.1 Verify an OPERABLE door is closed in the affected air lock. AND	 1 hour

ACTION a.1
DOC L02
ACTION a.2
DOC L01
DOC L01

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.2 Lock an OPERABLE door closed in the affected air lock. AND	24 hours
	B.3 ----------NOTE---------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1. AND	Immediately
	C.2 Verify a door is closed in the affected air lock. AND	1 hour
	C.3 Restore air lock to OPERABLE status.	24 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~ 3
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND	6 hours

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a, Action b	D.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	----------NOTES---------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	~~[~~ ~~24 months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

DOC A05

4.6.1.3.a
4.6.1.1.b

4.6.1.3.b

3

CTS

1

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks ~~(Atmospheric and Dual)~~ 1

3.6.1.3 LCO 3.6.2 ~~[~~Two~~]~~ containment air lock~~[s]~~ shall be OPERABLE. 3

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

----------NOTES----------

DOC L01 1. Entry and exit is permissible to perform repairs on the affected air lock components.

DOC A03 2. Separate Condition entry is allowed for each air lock.

ACTION NOTE 3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a A. One or more containment air locks with one containment air lock door inoperable.	----------NOTES---------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls ~~[~~if both air locks are inoperable~~]~~. --- A.1 Verify the OPERABLE door is closed in the affected air lock. <u>AND</u>	DOC L02 3 DOC M01 1 hour

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a.1	A.2 Lock the OPERABLE door closed in the affected air lock. AND	24 hours
DOC L02	A.3 ----------NOTE---------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days
ACTION a.2		
DOC L01 B. One or more containment air locks with containment air lock interlock mechanism inoperable.	----------NOTES---------- 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. ---------- B.1 Verify an OPERABLE door is closed in the affected air lock. AND	1 hour
DOC L01		

CTS

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.2 Lock an OPERABLE door closed in the affected air lock. AND B.3 ----------NOTE---------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- Verify an OPERABLE door is locked closed in the affected air lock.	24 hours Once per 31 days
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1. AND C.2 Verify a door is closed in the affected air lock. AND C.3 Restore air lock to OPERABLE status.	Immediately 1 hour 24 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND	6 hours

DOC L01 DOC L01 DOC L01 ACTION b DOC M02 DOC M01 ACTION b ACTION a, ACTION b

3

2

CTS

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a, Action b	D.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	----------NOTES---------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ----------	
DOC A05		
4.6.1.3.a 4.6.1.1.b	Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
4.6.1.3.b SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	~~[24 months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

3

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.2, CONTAINMENT AIR LOCKS

1. The type of Containment (Atmospheric and Dual) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ITS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks ~~(Atmospheric and Dual)~~ 1

BASES

BACKGROUND

Containment air locks (i.e., personnel and emergency) form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, approximately 9 ~~10~~ ft in diameter for the personnel air lock and approximately 6 ft in diameter for the emergency air lock, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door). 2

~~Each personnel air lock is provided with limit switches on both doors that provide control room indication of door position. Additionally, c~~ C ontrol room indication is provided to alert the operator whenever a ~~n~~ personnel air lock ~~door interlock mechanism is defeated~~ equalizing valve is open. 2

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

~~[For atmospheric containment, the DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), a main steam line break (MSLB) and a control element assembly (CEA) ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.10]% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option A (Ref. 1), as~~ 3

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~L_a: the maximum allowable containment leakage rate at the calculated [maximum] peak containment pressure (P_a) of [55.7] psig, which results from the limiting design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.~~ 3

For dual containment, the DBAs that result in a release of radioactive material within containment are a LOCA, an MSLB, and a CEA ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[~~0.50~~]~~% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option ~~A~~ B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated ~~[maximum]~~ peak containment pressure (P_a) of ~~[42.3]~~ 42.77 psig, which results from the limiting design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.~~]~~ 3 2 3

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable because of the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions. A third Note has been included that requires entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage limit.

A.1, A.2, and A.3

With one air lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

BASES

ACTIONS (continued)

In addition, the affected air lock penetration must be isolated by locking closed an OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

BASES

ACTIONS (continued)

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

BASES

ACTIONS (continued)

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock. 3

D.1 and D.2

~~---------- REVIEWER'S NOTE ----------~~
~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~
~~----------~~ 4

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is

BASES

ACTIONS (continued)

not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit into and out of containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment airlock door is used for entry and exit (procedures require strict adherence to single door opening), this test is ~~only required to be performed every 24 months~~ performed in accordance with the Surveillance Frequency Control Program. 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the airlock.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

REFERENCES

1. 10 CFR 50, Appendix J, Option ~~[A]~~[B~~]~~.
2. FSAR, Section ~~[~~ ~~]~~.
3. FSAR, Section ~~[~~ ~~]~~.
4. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks ~~(Atmospheric and Dual)~~

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, ~~10~~ ft in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches ~~on both~~ door~~s~~ that provide control room indication of door position. ~~Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated.~~

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

~~[For atmospheric containment, the DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), a main steam line break (MSLB) and a control element assembly (CEA) ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.10]% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option A (Ref. 1), as~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~L_a: the maximum allowable containment leakage rate at the calculated [maximum] peak containment pressure (P_a) of [55.7] psig, which results from the limiting design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.~~ 3

For dual containment, the DBAs that result in a release of radioactive material within containment are a LOCA, an MSLB, and a CEA ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[~~0.50~~]~~% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option ~~A~~ B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated ~~[maximum]~~ peak containment pressure (P_a) of ~~[42.3]~~ 43.43 psig, which results from the limiting design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.~~]~~ 3 2 3

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable because of the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions. A third Note has been included that requires entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage limit.

A.1, A.2, and A.3

With one air lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

BASES

ACTIONS (continued)

In addition, the affected air lock penetration must be isolated by locking closed an OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

BASES

ACTIONS (continued)

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

BASES

ACTIONS (continued)

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

~~---------- REVIEWER'S NOTE ----------~~
~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~
~~---------- ----------~~

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is

BASES

ACTIONS (continued)

not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit into and out of containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment airlock door is used for entry and exit (procedures require strict adherence to single door opening), this test is ~~only required to be performed every 24 months~~ performed in accordance with the Surveillance Frequency Control Program. 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the airlock.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

REFERENCES

1. 10 CFR 50, Appendix J, Option ~~[A]~~[B].
2. FSAR, Section ~~[ ]~~. 15
3. FSAR, Section ~~[ ]~~. 6.2

U 3 5 2

4. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.2 BASES, CONTAINMENT AIR LOCKS

1. The type of Containment (Atmospheric and Dual) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS Bases contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. The PSL Safety Analysis Report (SAR) is an updated version of the original Final Safety Analysis Report. Therefore, the proper acronym is UFSAR and is changed to reflect the document.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.6.2, CONTAINMENT AIR LOCKS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

ITS 3.6.3, CONTAINMENT ISOLATION VALVES

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

LCO 3.6.3

3.6.3.1 The containment isolation valves shall be OPERABLE:

Applicability

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Add proposed ACTIONS NOTE 2

A02

NOTE

ACTIONS NOTE 3

1. Enter applicable ACTIONS for systems made inoperable by containment isolation valves.

ACTIONS NOTE 4

2. Enter the ACTION of LCO 3.6.1.2, “Containment Leakage,” when leakage results in exceeding overall containment leakage rate acceptance criteria.

Add proposed ACTION D

L08

With one or more of the isolation valve(s) inoperable, either:

a. ~~Restore the inoperable valve(s) to OPERABLE status within 4 hours or in accordance with the Risk Informed Completion Time Program, or~~

A03

Required Actions A.1, B.1, and C.1

b. Isolate each affected penetration within 4 hours or in accordance with the Risk Informed Completion Time Program by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours or in accordance with the Risk Informed Completion Time Program by use of at least one closed manual valve or blind flange; or

or check valve with flow through the valve secured

L02

MODE 3

MODE 4

ACTION E

d. Be in at least ~~HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 6~~ hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION E NOTE

12

Add Completion Time of 1 hour for ACTION B

M03

Add Completion Time of 72 hours for ACTION C

L01

SURVEILLANCE REQUIREMENTS

~~4.6.3.1.1 The isolation valves shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of the cycling test, and verification of isolation time.~~

L03

Add proposed SR 3.6.3.1

M01

Add proposed SR 3.6.3.5

M04

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE ~~during the COLD SHUTDOWN or REFUELING MODE~~ in accordance with the Surveillance Frequency Control Program by:

L04

SR 3.6.3.6

a. Verifying that on an actual or simulated actuation ~~a Containment Isolation test signal, and/or SIAS test~~ signal, each isolation valve actuates to its isolation position.

L05

LA01

that is not locked, sealed, or otherwise secured in position,

L06

SR 3.6.3.4

4.6.3.1.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to the INSERVICE TESTING PROGRAM.

~~Pages 3/4 6-21 through 3/4 6-22 have been DELETED.~~

~~Page 3/4 6-23 is the next valid page.~~

~~DELETED~~

~~DELETED~~

DELETED

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT VESSEL

CONTAINMENT VESSEL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be maintained.

See ITS 3.6.1

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

ACTION B / ACTION E

Without CONTAINMENT VESSEL INTEGRITY, restore CONTAINMENT VESSEL INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in ~~COLD~~ ~~SHUTDOWN~~ within ~~the following 30~~ hours.

MODE 4

L10

12

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be demonstrated:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

Add proposed action Required Actions A.2 and C.2 NOTEs 1 and 2; and SRs 3.6.3.2 and SR 3.6.3.3 NOTE

L07

Required Actions A.2 and C.2

SR 3.6.3.2

SR 3.6.3.3

1. All containment vessel penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open on an intermittent basis under administrative control, and

ACTIONS NOTE 1

, or check valve with flow through the valve secured

L02

2. All containment vessel equipment hatches are closed and sealed.

See ITS 3.6.1

b. By verifying that each containment vessel air lock is OPERABLE per Specification 3.6.1.3

See ITS 3.6.2

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and outside

L07

SR 3.6.3.3

* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

DEFINITIONS

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

See ITS Chapter 1.0

CONTAINMENT VESSEL INTEGRITY

1.7 CONTAINMENT VESSEL INTEGRITY shall exist when:

a. All containment vessel penetrations required to be closed during accident conditions are either:

LCO 3.6.3

1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed position except for valves that are open on an intermittent basis under administrative control.

b. All containment vessel equipment hatches are closed and sealed,

See ITS Chapter 3.6.1

c. Each containment vessel air lock is in compliance with the requirements of Specification 3.6.1.3,

See ITS Chapter 3.6.2

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

See ITS Chapter 3.6.1

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be the seal water flow supplied from the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include shared (4 fingered) control element assemblies (CEAs) withdrawn into the upper guide structure (UGS) or evolutions performed with the UGS in place such as CEA latching/unlatching or verification of latching/unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

See ITS Chapter 1.0

CORE OPERATING LIMITS REPORT (COLR)

1.9a THE COLR is the unit-specific document that provides cycle specific parameter limits for the current operating reload cycle. These cycle-specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these limits is addressed in individual Specifications.

ADMINISTRATIVE CONTROLS

(2) conform to the guidance of Appendix I to 10 CFR Part 50, and
(3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

See ITS 5.5

h. Containment Leakage Rate Testing Program

A program to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program is in accordance with the guidelines contained in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 8, 2005 Type A test shall be performed no later than December 8, 2020.

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 42.8 psig. The containment design pressure is 44 psig.

The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests, $\leq$ 0.75 L_a for Type A tests, and $\leq$ 0.096 L_a for secondary containment bypass leakage paths.

See ITS 5.5

SR 3.6.3.7

Verify the combined leakage rate when pressurized to $\geq$ 42.77 psig

R1

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a.

2) For the personnel air lock door seal, leakage rate is < 0.01 L_a when pressurized to $\geq$ 1.0 P_a.

3) For the emergency air lock door seal, leakage rate is < 0.01 L_a when pressurized to $\geq$ 10 psig.

See ITS 5.5

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

LCO 3.6.3 3.6.3 The containment isolation valves shall be OPERABLE.

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTION:

Add proposed ACTIONS NOTE 2

A02

NOTE

- Enter applicable ACTIONS for systems made inoperable by containment isolation valves.
- Enter the ACTION of LCO 3.6.1.2, “Containment Leakage,” when leakage results in exceeding overall containment leakage rate acceptance criteria.

ACTIONS NOTE 3

ACTIONS NOTE 4

With one or more of containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

Add proposed ACTION D

L08

a. ~~Restore the inoperable valve(s) to OPERABLE status within 4 hours or in accordance with the Risk Informed Completion Time Program, or~~

A03

Required Action A.1, B.1, and C.1

b. Isolate each affected penetration within 4 hours or in accordance with the Risk Informed Completion Time Program by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours or in accordance with the Risk Informed Completion Time Program by use of at least one closed manual valve or blind flange; or

or check valve with flow through the valve secured

L02

MODE 3

MODE 4

ACTION F

d. Be in at least ~~HOT STANDBY~~ within the next 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 6~~ 12 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION F NOTE

Add Completion Time of 1 hour for ACTION B

M03

Add Completion Time of 72 hours for ACTION C

L01

SURVEILLANCE REQUIREMENTS

~~4.6.3.1 The containment isolation valves shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.~~

L03

Add proposed SR 3.6.3.2

M02

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each automatic containment isolation valve shall be demonstrated OPERABLE ~~during the COLD SHUTDOWN or REFUELING MODE~~ in accordance with the Surveillance Frequency Control Program by: L04

an actual or simulated actuation L05

a. Verifying that on ~~a Containment Isolation test signal (CIAS) and/or a Safety Injection test signal (SIAS)~~, each isolation valve actuates to its isolation position. LA01

SR 3.6.3.7

that is not locked, sealed, or otherwise secured in position, L06

b. Verifying that on ~~a Containment Radiation-High test~~ signal, each containment purge valve actuates to its isolation position. LA01

an actual or simulated actuation L05

SR 3.6.3.5 4.6.3.3 The isolation time of each power-operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to the INSERVICE TESTING PROGRAM.

~~Pages 3/4 6-22 through 3/4 6-23 have been DELETED.~~

~~Page 3/4 6-24 is the next valid page.~~

~~DELETED~~

~~DELETED~~

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

See ITS 3.6.1

APPLICABILITY: MODES 1*, 2*, 3, and 4.

ACTION:

ACTION B
ACTION F

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following 30~~ hours.

MODE 4

L10

12

Add proposed action Required Actions A.2 and C.2 NOTEs 1 and 2; and SRs 3.6.3.3 and SR 3.6.3.4 NOTE

L07

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

Required Actions A.2 and C.2
SR 3.6.3.3
SR 3.6.3.4

a. In accordance with the Surveillance Frequency Control Program by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open on an intermittent basis under administrative control.

ACTIONS NOTE 1

, or check valve with flow through the valve secured

L02

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

See ITS 3.6.2

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* In MODES 1 and 2, the RCB polar crane shall be rendered inoperable by locking the power supply breaker open.

See ITS 3.6.1.1

SR 3.6.3.4

** Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

L07

and outside

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.6.3

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

SR 3.6.3.1

a. Each 48-inch containment purge supply and exhaust isolation valve shall be sealed closed.

SR 3.6.3.2

b. The 8-inch containment purge supply and exhaust isolation valves may be open for purging and/or venting as required for safety related purposes such as:

1. Maintaining containment pressure within the limits of Specification 3.6.1.4.

2. Reducing containment atmosphere airborne radioactivity and/or improving air quality to an acceptable level for containment access.

Applicability

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

ACTION A

ACTION F

a. With a 48-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, close and/or seal close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following 30~~ hours.

MODE 4

ACTION A

ACTION F

b. With an 8-inch containment purge supply and/or exhaust isolation valve(s) open for reasons other than those stated in Specification 3.6.1.7.b, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following 30~~ hours.

12

MODE 4

L10

ACTION E

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Surveillance Requirements 4.6.1.7.3 and/or 4.6.1.7.4, within 24 hours ~~either restore the inoperable valve(s) to OPERABLE status or~~ isolate the affected penetration flow path by use of at least

A03

ACTION E.1

one closed and de-activated automatic valve with resilient seals or blind flange,

ACTION E.2

verify the affected penetration flowpath is isolated, and perform Surveillance

ACTION E.3

Requirement 4.6.1.7.3 or 4.6.1.7.4 for resilient seated valves closed to isolate the penetration flowpath, otherwise be in at least HOT STANDBY within the next

ACTION F

6 hours and in ~~COLD SHUTDOWN~~ within ~~the following 30~~ hours.

12

MODE 4

L10

ACTION E.3

1. Closed and de-activated automatic valve(s) with resilient seals used to isolate the penetration flowpath(s) shall be tested in accordance with either Surveillance Requirement 4.6.1.7.3 for 48-inch valves at least once per 6 months or Surveillance Requirement 4.6.1.7.4 for 8-inch valves at least once per 92 days.

ACTION E.2
NOTES 1 and 2

NOTE

Verification of isolation devices by administrative means is acceptable when they are located in high radiation areas or they are locked, sealed, or otherwise secured by administrative means.

R1

ACTION E.2

2. Verify the affected penetration flowpath is isolated once per 31 days following isolation for isolation devices outside containment and prior to entering MODE 4 from MODE 5 for isolation devices inside containment if not performed within the previous 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1 4.6.1.7.1 Each 48-inch containment purge supply and exhaust isolation valve shall be verified to be sealed-closed in accordance with the Surveillance Frequency Control Program.

except for one purge valve in a penetration flow path while in Condition E of this LCO

L09

~~4.6.1.7.2 Documentation shall be reviewed in accordance with the Surveillance Frequency Control Program to confirm that purging and venting were performed in accordance with Specification 3.6.1.7.b.~~

L11

SR 3.6.3.6 4.6.1.7.3 In accordance with the Surveillance Frequency Control Program each sealed closed 48-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying ~~that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a~~.

SR 3.6.3.6 4.6.1.7.4 In accordance with the Surveillance Frequency Control Program, each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying ~~that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a~~.

LA02

DEFINITIONS

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT VESSEL INTEGRITY

See ITS Chapter 1.0

1.7 CONTAINMENT VESSEL INTEGRITY shall exist when:

a. All containment vessel penetrations required to be closed during accident conditions are either:

LCO 3.6.3

1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open on an intermittent basis under administrative control.

b. All containment vessel equipment hatches are closed and sealed,

See ITS Chapter 3.6.1

c. Each containment vessel air lock is in compliance with the requirements of Specification 3.6.1.3,

See ITS Chapter 3.6.2

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

See ITS Chapter 3.6.1

1.8 CONTROLLED LEAKAGE shall be the seal water flow supplied from the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include evolutions performed with the upper guide structure (UGS) in place such as control element assembly (CEA) latching/unlatching or verification of latching/unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

1.9a THE COLR is the unit-specific document that provides cycle specific parameter limits for the current operating reload cycle. These cycle-specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these limits is addressed in individual Specifications.

See ITS Chapter 1.0

ADMINISTRATIVE CONTROLS (Continued)

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests, $\leq$ 0.75 L_a for Type A tests, and $\leq$ 0.096 L_a for secondary containment bypass leakage paths.

See ITS 5.5

SR 3.6.3.8

Verify the combined leakage rate when pressurized to $\geq$ 43.43 psig

R1

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage is $\leq$ 0.05 L_a when tested at $\geq$ P_a.

2) For each door seal, leakage rate is < 0.01 L_a when pressurized to $\geq$ P_a.

See ITS 5.5

The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.

The provisions for T.S. 4.0.3 are applicable to the Containment Leak Rate Testing Program.

i. Deleted

DISCUSSION OF CHANGES
ITS 3.6.3, CONTAINMENT ISOLATION VALVES

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.6.3.1 and Unit 2 3.6.3 ACTIONS provide requirements to be taken for each inoperable containment isolation valve. ITS 3.6.3 includes an explicit Note (ACTION Note 2) that states, “Separate Condition entry is allowed for each penetration flow path.” This Note provides instructions for the proper application of the ACTIONS for ITS compliance. This changes the CTS by providing explicit direction for using the ACTIONS when a containment isolation valve is inoperable.

This change is acceptable because the addition of the Note reflects the CTS allowance to take appropriate Actions. This change is designated as administrative since it does not result in a technical change to the CTS.

A03 When one or more containment isolation valves are inoperable, Unit 1 CTS 3.6.3.1 and Unit 2 CTS 3.6.3 Action a requires restoring the inoperable valve(s) to OPERABLE status within 4 hours or taking one of the other specified compensatory actions. In addition, Unit 2 CTS 3.6.1.7 Action c requires restoring the inoperable valve(s) to OPERABLE status. ITS 3.6.3 does not state the requirement to restore an inoperable isolation valve to OPERABLE status but includes other compensatory Required Actions to take within in 4 hours or 72 hours, as applicable. This changes the CTS by not explicitly stating the requirement to restore an inoperable valve to OPERABLE status.

This change is acceptable because the technical requirements have not changed. Restoration of compliance with the LCO is always an available action and it is the convention in the ITS to not state such “restore” options explicitly unless it is the only action or is required for clarity. This change is designated as administrative because it does not result in any technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS 3.6.3.1 does not have specific requirements for the 48 inch purge valves. ITS SR 3.6.3.1 requires verification that each 48 inch purge valve is sealed closed in accordance with the Surveillance Frequency Control Program except for purge valves in a penetration flow path while in Condition D. This changes the CTS by adding a specific Surveillance Requirement to verify each 48 inch purge valve is sealed closed in MODES 1, 2, 3, and 4. Actions are also added for excessive leakage in containment penetration flow paths containing purge valves (Refer to Discussion of Change (DOC) L08).

The purpose of ITS SR 3.6.3.1 is to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a 48 inch containment purge valve. A sealed closed valve must have motive power to the valve operator removed. The initial Frequency in accordance with the Surveillance Frequency Control Program will be specified as 31 days consistent with ISTS 3.6.3.1 and based on an NRC initiative, Generic Issue B-24, "Containment Leakage Due to Seal Deterioration." This change is acceptable and is designated as more restrictive since the CTS does not currently require this verification for the 48 inch purge valves.

M02 **Unit 2 only:** CTS 3.6.3 and CTS 3.6.1.7 do not have a specific surveillance requirement to verify each 8 inch purge valve is closed. However, CTS 3.6.1.7.b allows the 8 inch purge supply and exhaust isolation valves to be opened for purging and venting as required with restrictions. This implies that the 8 inch purge valves must be closed at other times during MODES 1, 2, 3 and 4. ITS SR 3.6.3.2 requires verification that each 8 inch purge valve is closed in accordance with the Surveillance Frequency Control Program except when the 8 inch purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. This changes the CTS by adding a specific Surveillance Requirement to verify each 8 inch purge valve is closed in MODES 1, 2, 3, and 4. |R1

The purpose of ITS SR 3.6.3.2 to ensure that the 8 inch purge valves are closed as required or, if open, it is for an allowable reason. The initial Frequency in accordance with the Surveillance Frequency Control Program will be specified as 31 days consistent with ISTS 3.6.3.2 and based on the Frequency of verifying the position of containment isolation valves. This change is acceptable and is designated as more restrictive since the CTS does not currently require this verification for the 8 inch purge valves.

M03 Unit 1 CTS 3.6.3.1 and Unit 2 CTS 3.6.3 Actions require, in part, with one or more isolation valve(s) inoperable to isolate the affected penetration. CTS Actions b and c allow 4 hours to isolate the affected penetration when one or more containment isolation valves are inoperable. ITS 3.6.3 Required Action B.1 requires 1 hour to isolate the affected penetration flow path when both valves in the same penetration flow path are inoperable. This changes the CTS by prescribing the time allowed to isolate the affected penetration when both containment isolation valves in the same penetration flow path are inoperable.

The purpose of the CTS Actions is to provide compensatory actions for inoperable containment isolation valves. However, when both valves in the same penetration flow path are inoperable, the time allowed to isolate the affected penetration flow path is not prescribed and action must be taken immediately to maintain at least one isolation valve operable. Because the containment isolation valves support the leak tightness of the containment the time allowed to isolate the affected penetration should be the same as that allowed to restore an inoperable containment. The one hour Completion Time is commensurate with the Completion Time in ITS 3.6.1 (CTS 3.6.1.1 (U1) and 3.6.3 (U2)) for restoring the containment to OPERABLE status when the containment is inoperable. This change is considered more restrictive because it provides a new Condition and

associated Required Actions for two inoperable containment isolation valves in the same containment penetration flow path.

M04 **Unit 1 only:** CTS 3.6.3.1 does not have a specific surveillance requirement to perform leakage rate testing for containment purge valves with resilient seals. ITS SR 3.6.3.5 requires leakage rate testing for containment penetration flow paths containing purge valves with resilient seals in accordance with the Surveillance Frequency Control Program. This changes the CTS by adding a specific Surveillance Requirement to perform leakage rate testing for containment penetration flow paths containing purge valves with resilient seals. Actions are also added for excessive leakage in containment penetration flow paths containing purge valves (Refer to DOC L08).

The purpose of ITS SR 3.6.3.5 to perform additional leakage rate testing beyond the 10 CFR 50, Appendix J requirements to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter period of time than do other seal types. The initial Frequency in accordance with the Surveillance Frequency Control Program will be specified as 184 days consistent with ISTS SR 3.6.3.6 and based on the Frequency established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration." This change is acceptable and is designated as more restrictive since the CTS does not currently require this Surveillance.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 4.6.3.1.2.a states that each containment isolation valve shall be demonstrated OPERABLE by verifying that on “a Containment Isolation test signal, and/or SIAS Test” signal actuates to its isolation position. Unit 2 CTS 4.6.3.2.a states that on “a Containment Isolation test signal (CIAS) and/or a Safety Injection test signal (SIAS)” actuates to its isolation position and Unit 2 CTS 4.6.3.2.b. states that on “a Containment Radiation High test signal” each containment purge valve actuates to its isolation position. Unit 1 ITS 3.6.3.6 and Unit 2 ITS SR 3.6.3.7 require verification that each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal. This changes the CTS by moving the detail concerning what type of signals are used to conduct the Surveillance Requirement to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to verify that the required valve automatically actuate. Also, this change is acceptable because the removed information will be adequately controlled in the

ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 **Unit 2 only:** *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.6.1.7.3 and 4.6.1.7.4 require the containment purge and exhaust valve leakage rate to be less than or equal to 0.05 La when pressurized to Pa. ITS SR 3.6.3.6 does not include this requirement. This changes the CTS by moving the specific acceptance criteria to the Bases.

The removal of these details, which provide specific leakage acceptance criteria for purge valve leakage, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS SR 3.6.3.6 retains the requirement to perform leakage rate testing for containment penetration flow paths containing purge valves with resilient seals. Also, this change is acceptable because these types of details will be adequately controlled by the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirement are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 3 – Relaxation of Completion Time)* Unit 1 CTS 3.6.3.1 and Unit 2 CTS 3.6.3 Actions b and c state that with one or more of the containment isolation valve(s) inoperable, isolate each affected penetration within 4 hours by use of one deactivated automatic valve secured in the isolation position, closed manual valve, or blind flange. ITS 3.6.3 ACTION C, which only applies to penetration flow paths with only one containment isolation valve and in a closed system, requires that with one or more penetration flow paths with one containment isolation valve inoperable, the penetration flow path be isolated by means similar to those specified in the CTS within 72 hours. This changes the CTS by extending the Completion Time from 4 hours to 72 hours when the inoperable containment isolation valve is in penetration flow path associated with a closed system. This also changes the Unit 2 CTS by providing an Action for a single valve penetration in a closed system, consistent with the Unit 1 CTS, instead of entering CTS 3.0.3.

The purpose of the CTS 3.6.3.1 Actions b and c is to provide a degree of assurance that the penetration flow path with an inoperable containment isolation valve maintains the containment penetration isolation boundary. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, the capacity and capability of remaining features (i.e., a closed system), a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed

Completion Time. In the case of a penetration flow path in a closed system with an inoperable valve, 72 hours is a reasonable time period considering the relative stability of the closed system to act as a penetration isolation boundary, or the small diameter of the pipe penetration and the instrument to act as a penetration isolation boundary. This change is designated as less restrictive because additional time is allowed to restore the components to within the LCO limits than was allowed in the CTS.

L02 *(Category 4 - Relaxation of Required Action)* Unit 1 CTS 3.6.3.1 and Unit 2 CTS 3.6.3 Action b. and c. state that with one or more of the containment isolation valve(s) inoperable, isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position (Action b), closed manual valve (Action c), or blind flange (Action c). In addition, Unit 1 CTS 4.6.1.1.a.1 and Unit 2 CTS 4.6.1.1.a require a periodic verification that the affected penetration remains isolated by the same methods. When one or more penetration flow paths with one containment isolation valve inoperable, ITS 3.6.3 Required Action A.1 requires that the affected penetration flow path be isolated by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. In addition, ITS 3.6.3 Required Action A.2 requires a periodic verification that the affected penetration remains isolated by one of the methods required by ITS 3.6.3 Required Action A.1. This changes the CTS by allowing penetration flow paths with two containment isolation valves that have one containment isolation valve inoperable to use a check valve with flow through the valve secured as the means of isolating the penetration flow path.

The purpose of Unit 1 CTS 3.6.3.1 Unit 2 CTS 3.6.3 b and c actions, and Unit 1 CTS 4.6.1.1.a.1 and Unit 2 CTS 4.6.1.1.a Surveillances is to provide assurance that the affected penetration flow path is isolated. This change is acceptable. because the ITS Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The ITS Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. This change allows the flow path to be isolated by one check valve with flow through the valve secured. The requirement to isolate the flow path is retained and using a check valve with flow through the valve secured is an appropriate method of isolation. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 5 – Deletion of Surveillance Requirement)* Unit 1 CTS 4.6.3.1.1 and Unit 2 CTS 4.6.3.1 describe tests that must be performed prior to returning a valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit. The ITS does not include these testing requirements. This changes the CTS by deleting this post-maintenance Surveillance.

The purpose of Unit 1 CTS 4.6.3.1.1 and Unit 2 CTS 4.6.3.1 is to verify OPERABILITY of containment isolation valves following their maintenance, repair, or replacement. This change is acceptable because the deleted Surveillance Requirement is not necessary to ensure that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. Any time the OPERABILITY of a system or component has been affected by repair, maintenance, modification, or replacement of a component, post-maintenance testing is required to demonstrate the OPERABILITY of the system or component. This is described in the Bases for ITS SR 3.0.1 and required under SR 3.0.1. The OPERABILITY requirements for the containment isolation valves are described in the Bases for ITS 3.6.3. In addition, the requirements of 10 CFR 50, Appendix B, Section XI (Test Control), provide adequate controls for test programs to ensure that testing incorporates applicable acceptance criteria. Compliance with 10 CFR 50, Appendix B, is required under the unit operating license. As a result, post-maintenance testing will continue to be performed and an explicit requirement in the Technical Specifications is not necessary. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L04 *(Category 7 – Relaxation Of Surveillance Frequency Change)* Unit 1 CTS 4.6.3.1.2 and Unit 2 CTS 4.6.3.2 state, in part, that each containment isolation valve shall be demonstrated Operable during Cold Shutdown or Refueling Mode in accordance with the Surveillance Frequency Control Program by verifying the valve actuates on an isolation signal. Unit 1 ITS SR 3.6.3.6 and Unit 2 ITS SR 3.6.3.7 require verifying each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated signal without requiring the verification be performed in cold shutdown or refueling modes. This changes the CTS by removing the restriction on surveillance performance during specific Modes.

The purpose of the CTS Surveillance is to demonstrate Operability of automatic containment isolation valves. The proposed change is acceptable because it does not change the method of test or frequency of the affected surveillances. The proposed change only deletes the requirement to perform this testing during shutdown or refueling conditions. In addition, allowing this testing to be performed either at refueling, shutdown or at power does not affect the applicable safety analysis conclusions and allows shutdown activities to be planned which will help reduce risk and Increase equipment availability during shutdowns. Thus, the proposed change will continue to provide adequate assurance the required components are routinely tested to ensure system operability while providing some additional flexibility in planning and scheduling the required testing. In addition, due to system designs that allow for safe testing at power, the proposed change will not adversely affect the safe operation of the plant. The proposed change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L05 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.6.3.1.2.a and Unit 2 CTS 4.6.3.2.a. and b. require verification of the

containment isolation valve actuation on specific isolation test signals. ITS SR 3.6.3.6 and Unit 2 ITS SR 3.6.3.7 specify that the signal may be either an "actual" or a simulated actuation signal. This changes the Unit 1 and Unit 2 CTS by allowing the use of either an actual or a simulated signal for the test.

The purpose of CTS Surveillances is to ensure that the containment isolation valves operate correctly upon receipt of an actuation signal. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual," "simulated," or "test" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L06 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.6.3.1.2.a and CTS 4.6.1.1.a.1, and Unit 2 CTS 4.6.3.2.a. and 4.6.1.1.a require verification that each containment isolation valve actuates to its isolation position. Unit 1 ITS SR 3.6.3.6 and Unit 2 ITS SR 3.6.3.7 require verification that each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal. This changes the CTS by not requiring automatic valves locked, sealed or otherwise secured in position to be tested to verify that they automatically actuate to their isolation position.

The purpose of CTS Surveillances is to provide assurance that the automatic valves required to actuate in case of a design basis accident (DBA) isolate containment properly. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Automatic valves already in the isolated position and secured are not required to be tested to automatically actuate because, in case of a DBA, they are already in their required position and secured to prevent changing from the required position. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L07 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.6.1.1.a. requires verification that specified containment penetrations are closed. These CTS requirements also contain a note (* Note for Unit 1 and **Note for Unit 2) that exempts testing for valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. ITS 3.6.3 Required Actions A.2 and C.2, Unit 1 ITS SRs 3.6.3.2 and 3.6.3.3 and Unit 2 SRs 3.6.3.3 and 3.6.3.4 include similar requirements but contain a Note that allows valves and blind flanges in high radiation areas to be verified administratively and except valves, whether inside containment or outside containment, that are required to be closed in an accident and are locked, sealed or otherwise secured in the closed position. In addition, ITS 3.6.3 Required Actions A.2 and C.2

include a second Note that allows verification of isolation devices that are locked, sealed, or otherwise secured to also be performed using administrative means. This changes the CTS by allowing certain valves and blind flanges to not require physical verification.

The purpose of CTS 4.6.1.1.a is to provide assurance that containment penetrations are closed when necessary. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. The position of containment isolation valves and blind flanges in high radiation areas that are required to be closed can be verified administratively, not requiring physical verification. Access to high radiation areas is limited, making access to the valves and blind flanges more difficult, and mispositioning less likely. For those isolation devices, including devices outside containment, that are locked, sealed, or otherwise secured, plant procedures control their operation. Therefore, the potential for inadvertent misalignment of these devices after locking, sealing, or securing is low. In addition, all the isolation devices were verified to be in the correct position (as required by ITS 3.6.3 Required Actions A.2 and C.2) prior to locking, sealing, or otherwise securing. This change is designated as less restrictive because less stringent Surveillance Requirements and Required Actions are being applied in the ITS than were applied in the CTS.

L08 *(Category 3 – Relaxation of Completion Time)* Unit 1 CTS 3.6.3.1 does not have an explicit ACTION for secondary containment bypass leakage or containment penetration flow paths with purge valve leakage not within limits. Unit 2 CTS 3.6.3 does not have an explicit ACTION for secondary containment bypass leakage. Because CTS do not explicitly address these leakage limits, CTS 3.6.1.2 ACTION states, “With the containment leakage rate exceeding the acceptance criteria of the Containment Leakage Rate Testing Program, within 1 hour initiate action to be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.” Unit 1 ITS ACTION D requires secondary bypass leakage to be restored within limit in 4 hours and purge valve leakage to be restored within 24 hours. Unit 2 ITS ACTION D requires secondary bypass leakage to be restored within limit in 4 hours and Unit 2 ACTION E requires isolation of the affected penetration flow path when purge valve leakage is not within leakage limits. This changes the CTS by adding an explicit Action time to restore secondary containment bypass and purge valve leakage to within the leakage limits rather than requiring the unit be in Cold Shutdown within 36 hours.

The purpose of Unit 1 and Unit 2 ITS 3.6.3 ACTION D is to restore secondary containment bypass leakage to within the assumptions of the safety analysis. This change is acceptable because the 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating penetration(s) and the relative importance of secondary containment bypass leakage to the overall containment function. The Unit 1 Required Action D.1 24-hour Completion Time for purge valve leakage is acceptable considering the purge valves remain closed so that a gross breach of containment does not exist. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L09 **Unit 2 only:** *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.6.1.7.1 requires verification that each 48 inch containment purge supply and exhaust isolation valve be verified to be sealed closed in accordance with the Surveillance Frequency Control Program. ITS SR 3.6.3.1 provides the exception for purge valves in a penetration flow path while in Condition E of the LCO. This changes the CTS by not requiring the Surveillance Requirement to be met while in Condition E.

The purpose of CTS 4.6.1.7.1 is to ensure that a gross breach of the containment is not caused by an inadvertent or spurious opening of a 48 inch containment purge valve. The change is acceptable because the relaxed Surveillance Requirement acceptance criteria detail is not necessary for verification that the penetration can meet its required function. The Surveillance Requirement is not required to be met while in Condition E since the penetration flow path would be isolated and verified isolated periodically. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L10 *(Category 4 - Relaxation of Required Action)* CTS 3.6.1.1 requires the plant to be in at least MODE 3 in 6 hours and in MODE 5 within the following 30 hours when containment vessel penetrations not capable of being closed by OPERABLE containment isolation valves required to be closed during accident conditions are isolated by other allowed means (CTS 4.6.1.1).

Unit 2 CTS 3.6.1.7 Actions a, b and c require the plant to be in at least MODE 3 in 6 hours and in MODE 5 within the following 30 hours if the 48 inch containment purge valves are open or not sealed closed and the penetration is not isolated within 4 hours, the 8 inch containment purge valves are open for reasons other than containment pressure control, air quality improvement for containment access or to reduce containment atmosphere airborne radioactivity and the penetration is not isolated within 4 hours; or, the containment purge valve measured leakage exceeds limits and the penetration is not isolated within 24 hours.

Unit 1 ITS 3.6.3 ACTION E and Unit 2 ITS 3.6.3 ACTION F require that the plant be in MODE 3 within 6 hours and MODE 4 within 12 hours. This changes the end state for CTS for the applicable condition.

This change is acceptable because the modification of the end state from MODE 5 to MODE 4 is consistent with CE-NPSD-1186-A, Revision 00, “Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs,” dated October 2001 (ADAMS Accession No. 110410539). This change is designated as less restrictive because instead of requiring the plant to achieve MODE 5 (COLD SHUTDOWN), the ITS end state is MODE 4 (HOT SHUTDOWN) with LCO 3.0.4.a allowances not applicable.

L11 **Unit 2 only:** (*Category 5 – Deletion of Surveillance Requirement*) CTS 4.6.1.7.2 states, “Documentation shall be reviewed in accordance with the Surveillance Frequency Control Program to confirm that purging and venting were performed in accordance with Specification 3.6.1.7.b.” ITS 3.6.3 does not include this

quality assurance detail. This changes the CTS by deleting a documentation review requirement.

The purpose of CTS 4.6.1.7.2 is to confirm that the 8 inch purge valves are opened for the purposes specified in Specification 3.6.1.7.b. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the purge valves are Operable and can perform their required functions of supporting the containment and containment isolation. The 8 inch purge supply and exhaust valves continue to be tested in a manner and at a frequency necessary to give confidence that they can perform their intended safety function. This change is acceptable because determining if an activity was properly performed after the activity has been completed is a quality assurance attribute and has no impact to the immediate compliance of the requirement while the activity is performed and, therefore, is unnecessary and not required to ensure the LCO met or the remedial actions performed, as required. The PSL quality assurance program requires that surveillance, inspection and test reports, routine testing and technical specification be controlled under the document control program as quality related products, which provides effective monitoring of activities of this type and provide for corrective actions, if necessary. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves ~~(Atmospheric and Dual)~~ 1

3.6.3.1 1.7 LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

8

----------NOTES----------

4.6.1.1.a.1 1. Penetration flow paths ~~[~~except for ~~[4~~2~~]~~ inch purge valve penetration flow paths~~]~~ may be unisolated intermittently under administrative controls. 3

DOC A02 2. Separate Condition entry is allowed for each penetration flow path.

3.6.3.1 ACTIONS NOTE 1 3. Enter applicable Conditions and Required Actions for system(s) made inoperable by containment isolation valves.

3.6.3.1 ACTIONS NOTE 2 4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. ----------NOTE---------- Only applicable to the [containment sump supply valves to the ECCS and containment spray pumps]. ---------- One or more penetration flow paths with one containment isolation valve inoperable.~~	~~A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.~~ ~~AND~~	~~4 hours [OR In accordance with the Risk Informed Completion Time Program]~~ 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~A.2 ----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ----------~~ ~~Verify the affected penetration flow path is isolated.~~	~~Once per 31 days [following isolation] for isolation devices outside containment~~ ~~AND~~ ~~Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment~~
~~[B~~. ----------NOTE---------- Only applicable to penetration flow paths with two ~~[or more]~~ containment isolation valves. ---------- One or more penetration flow paths with one containment isolation valve inoperable ~~[~~for reasons other than Condition~~[s] A, E, [and F]]~~.	~~B~~.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	~~[7 days]~~ ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~

DOC L02

3.6.3.1 Actions b. and c.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A ~~B~~.2 ----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days ~~[~~following isolation~~]~~ for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment~~]~~
B ~~C~~. ----------NOTE---------- Only applicable to penetration flow paths with two ~~[or more]~~ containment isolation valves. ---------- One or more penetration flow paths with two ~~[or more]~~ containment isolation valves inoperable ~~[~~for reasons other than Condition~~[s] E [and F]]~~. D	~~C~~.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

DOC L07

4.6.1.1.a.1

DOC M03

3.6.3.1 Actions b. and c.

3.6.1.1 Action

2 2 3 3 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D~~. ----------NOTE---------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ---------- One or more penetration flow paths with one containment isolation valve inoperable.	~~D~~.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours ~~for those penetrations that do not met the 7 day criteria~~ ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~ ~~AND~~ ~~7 days for those penetrations that meet the 7 day criteria~~ ~~[OR In accordance with the Risk Informed Completion Time Program]~~
	AND ~~D~~.2 ----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days ~~[~~following isolation~~]~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L08 D S ~~E~~. ~~[One or more s~~econdary containment bypass leakage ~~[or purge valve leakage]~~ not within limit.	D ~~E~~.1 Restore leakage within limit.	4 hours for secondary containment bypass leakage AND 24 hours for purge valve leakage ~~]~~
~~F. [One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.~~	~~F.1 Isolate the affected penetration flow path by use of at least one [closed and de-activated automatic valve with resilient seals, closed manual valve with resilient seals, or blind flange].~~ ~~AND~~	~~24 hours~~

OR

One or more containment penetration flow paths with purge valve leakage not within limits

2
3
3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~F.2 NOTES 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.~~	
	~~Verify the affected penetration flow path is isolated.~~	~~Once per 31 days for isolation devices outside containment~~ ~~AND~~ ~~Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment~~
	~~AND~~	
	~~F.3 Perform SR 3.6.3.6 for the resilient seal purge valves closed to comply with Required Action F.1.~~	~~Once per [ ] days~~]

3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~G~~. Required Action and associated Completion Time not met.	~~G~~.1 Be in MODE 3.	6 hours
	AND	
	~~G~~.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	~~[~~ Verify each ~~[42]~~ inch purge valve is sealed closed except for ~~one~~ purge valve in a penetration flow path while in Condition ~~E~~ of this LCO.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~
~~SR 3.6.3.2~~	~~Verify each [8] inch purge valve is closed except when the [8] inch purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.~~	~~[31 days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.~~3~~ (2)	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ----------	(2)
4.6.1.1.a.1	Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)
4.6.1.1.a SR 3.6.3.~~4~~ (3)	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ----------	(2)
4.6.1.1.a.1 Note *	Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days

DOC L07

DOC L07

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.6.3.1.3	SR 3.6.3.~~5~~ (4)	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	~~[~~In accordance with the INSERVICE TESTING PROGRAM ~~OR~~ ~~[92 days]~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~	(2) (3)
DOC M04	SR 3.6.3.~~6~~ (5)	Perform leakage rate testing for containment purge valves with resilient seals. penetration flow paths containing	~~[184 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ ~~AND~~ ~~Within 92 days after opening the valve~~	(2) (3) (4)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.~~7~~ (6)	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2) (3)
~~SR 3.6.3.8~~	~~[Verify each [ ] inch containment purge valve is blocked to restrict the valve from opening > [50]%.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~ (2)
SR 3.6.3.~~9~~ (7)	~~[~~ Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ ~~[~~ L_a ~~]~~ (0.096) when pressurized to ≥ ~~[~~ psig~~]~~ (42.77).	In accordance with the Containment Leakage Rate Testing Program~~]~~ (2) (3) R1

4.6.3.1.2.a
DOC L05
DOC L06
DOC LA01
DOC L04
6.8.4.h.a

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves ~~(Atmospheric and Dual)~~ 1

3.6.3
3.6.1.7
1.7

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

8

----------NOTES----------

4.6.1.1.a

1. Penetration flow paths ~~[~~except for ~~[4~~2~~]~~ inch purge valve penetration flow paths~~]~~ may be unisolated intermittently under administrative controls. 3

DOC A02

2. Separate Condition entry is allowed for each penetration flow path.

3.6.3 ACTIONS NOTE

3. Enter applicable Conditions and Required Actions for system(s) made inoperable by containment isolation valves.

3.6.3 ACTIONS NOTE

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. ----------NOTE---------- Only applicable to the [containment sump supply valves to the ECCS and containment spray pumps]. ---------- One or more penetration flow paths with one containment isolation valve inoperable.~~	~~A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.~~ AND	~~4 hours~~ [OR ~~In accordance with the Risk Informed Completion Time Program]~~ 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~A.2 ----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ----------~~ ~~Verify the affected penetration flow path is isolated.~~	~~Once per 31 days [following isolation] for isolation devices outside containment~~ ~~AND~~ ~~Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment~~
~~[B~~. ----------NOTE---------- Only applicable to penetration flow paths with two ~~[or more]~~ containment isolation valves. ---------- One or more penetration flow paths with one containment isolation valve inoperable ~~[~~for reasons other than Condition~~[s] A, E, [and F]]~~.	~~B~~.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	~~[7 days]~~ ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~

DOC L02

3.6.3 Actions b. and c.

3.6.1.7 Actions a. and b.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~B~~.2 A ----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days ~~[~~following isolation~~]~~ for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment~~]~~
~~C~~. B ----------NOTE---------- Only applicable to penetration flow paths with two ~~[or more]~~ containment isolation valves. ---------- One or more penetration flow paths with two ~~[or more]~~ containment isolation valves inoperable ~~[~~for reasons other than Condition~~[s]~~ ~~E~~ ~~[~~and ~~F]]~~. D E	~~C~~.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

DOC L07

4.6.1.1.a

3.6.3 Actions b. and c.

3.6.1.1 Action

DOC M03

2

3

2

3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C ~~D~~. ----------NOTE---------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ---------- One or more penetration flow paths with one containment isolation valve inoperable.	~~D~~.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours ~~for those penetrations that do not met the 7 day criteria~~ ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~ ~~AND~~ ~~7 days for those penetrations that meet the 7 day criteria~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~
	AND C ~~D~~.2 ----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days ~~[~~following isolation~~]~~

3.6.3 Actions b. and c.

DOC L01

DOC L07

4.6.1.1.a

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D S ~~E.~~ ~~[One or more s~~econdary containment bypass leakage ~~[or purge valve leakage] ~~not within limit.	D ~~E~~.1 Restore leakage within limit.	4 hours ~~for secondary containment bypass leakage~~ ~~AND~~ ~~24 hours for purge valve leakage]~~
E ~~F~~. ~~[~~One or more penetration flow paths with ~~one or more~~ containment purge valves not within ~~purge valve~~ leakage limits.	E ~~F~~.1 Isolate the affected penetration flow path by use of at least one ~~[~~closed and de-activated automatic valve with resilient seals, closed manual valve with resilient seals, or blind flange~~]~~. AND	24 hours

DOC L08

3.6.1.7.c

2 3

2 3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~F~~.2 E ----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ----------	
3.6.1.7.c.2	Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
3.6.1.7.c.1	AND ~~F~~.3 E Perform SR 3.6.3.6 for the resilient seal ~~purge~~ valves closed to comply with Required Action ~~F~~. 1. E	for 8 inch valves Once per 92 [~~]~~ days ~~]~~ AND Once per 6 months for 48 inch valves.

R1

2 3 2 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~G~~. Required Action and associated Completion Time not met.	~~G~~.1 Be in MODE 3.	6 hours
	AND	
	~~G~~.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	12 hours

DOC L10
3.6.3 ACTION d.
3.6.1.1 ACTION
3.6.1.7 Action a., b. and c.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	~~[~~ Verify each ~~[42]~~ inch purge valve is sealed closed except for ~~one~~ purge valve in a penetration flow path while in Condition E of this LCO.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~
SR 3.6.3.2	Verify each ~~[~~8~~]~~ inch purge valve is closed except when the ~~[~~8~~]~~ inch purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.6.1.7.1
3.6.1.7.a
DOC L09
3.6.1.7.b
DOC M02

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.3 DOC L07 3.6.1.7 Action c. NOTE 4.6.1.1.a	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 3
SR 3.6.3.4 DOC L07 3.6.1.7 Action c. NOTE 4.6.1.1.a Note **	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.6.3.3	SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	~~[~~In accordance with the INSERVICE TESTING PROGRAM ~~OR~~ ~~[92 days]~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~	3
4.6.1.7.3 4.6.1.7.4 3.6.1.7.c.1 DOC LA02	SR 3.6.3.6	Perform leakage rate testing for containment purge valves with resilient seals. penetration flow paths containing	~~[184 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ ~~AND~~ ~~Within 92 days after opening the valve~~	2 3 4

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
~~SR 3.6.3.8~~	~~[Verify each [ ] inch containment purge valve is blocked to restrict the valve from opening > [50]%.~~	~~[~~ ~~[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~
SR 3.6.3.~~9~~ (8)	~~[~~ Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ ~~[~~ L_a ~~]~~ (0.096) when pressurized to ≥ ~~[~~ psig~~]~~. (43.43)	In accordance with the Containment Leakage Rate Testing Program~~]~~

4.6.3.2.a
4.6.3.2.b
DOC L05
DOC L06
DOC LA01
DOC L04
6.8.4.h.a

3 2 2 3 R1

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.3, CONTAINMENT ISOLATION VALVES

1. The type of Containment (Atmospheric and Dual) is deleted since it is unnecessary (only one Containment Specification is used in the St. Lucie Plant (PSL) Unit 1 and 2 ITS). This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ITS which reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. ISTS SR 3.6.3.6 requires performance of leakage rate testing for containment penetration flow paths containing purge valves with resilient seals within 92 days after opening a valve. Unit 2 ITS SR 3.6.3.6 deletes the 92 day verification requirement based on Unit 2 current licensing basis. The Surveillance is performed in accordance with the Surveillance Frequency Control Program as implemented per Amendment No. 173 (ADAMS Accession No. ML15127A066). Unit 1 ITS SR 3.6.3.5 has been added as a more restrictive requirement (See DOC M04) that deletes the 92 day verification requirement consistent with the PSL Unit 2 Surveillance. The Surveillance is performed at a frequency in accordance with the Surveillance Frequency Control Program.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves ~~(Atmospheric and Dual)~~ 1

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on an automatic isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. One of these barriers may be a closed system.

, Safety Injection Actuation Signal

containment high radiation

Containment isolation occurs upon receipt of a high containment pressure signal or a ~~low Reactor Coolant System (RCS) pressure~~ signal. The containment isolation signal closes automatic containment isolation valves in fluid penetrations not required for operation of Engineered Safety Feature systems in order to prevent leakage of radioactive material. Upon actuation of safety injection, automatic containment isolation valves also isolate systems not required for containment or RCS heat removal. Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated in the event of a release of radioactive material to containment atmosphere from the RCS following a Design Basis Accident (DBA). 2

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the accident analysis will be maintained.

~~The purge valves were designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor RCS leakage prior to personnel entry into containment. There are two sets of purge valves: normal purge and exhaust valves and minipurge and exhaust valves.~~ The ~~normal and minipurge~~ supply and exhaust lines are each supplied with inside and outside containment isolation valves but share common supply and exhaust penetration lines. 2

BASES

BACKGROUND (continued)

to provide a means of removing airborne radioactivity following operation of the containment airborne radioactivity removal system and

The ~~normal~~ purge valves are designed ~~for~~ purging the containment atmosphere to the unit stack while introducing filtered makeup from the outside to provide adequate ventilation for personnel comfort when the unit is shut down during refueling operations and maintenance. ~~Motor operated isolation valves are provided inside the containment, and air operated isolation valves are provided outside the containment.~~ The valves are operated manually from the control room. The valves will close automatically upon receipt of a containment ~~purge~~ isolation signal~~.~~ ~~The air operated valves~~ fail closed upon a loss of air. Because of their large size, the ~~normal~~ purge valves ~~in some units~~ are not qualified for automatic closure from their open position under DBA conditions. Therefore, the ~~normal~~ purge valves are normally maintained closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Air operated isolation valves are provided both inside and outside the containment.

and

2

Open ~~normal~~ purge valves~~, or a failure of the minipurge valves to close,~~ following an accident that releases contamination to the containment atmosphere would cause a significant increase in the containment leakage rate. 2

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), a main steam line break, and a control element assembly ejection accident. In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analysis assumes that the ~~normal~~ purge valves are closed at event initiation. 2

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The single failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. ~~The inboard and outboard isolation valves on each line are provided with diverse power sources, motor operated and pneumatically operated spring closed, respectively.~~ This arrangement was designed to preclude common mode failures from disabling both valves on a purge line. 2

The purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valves due to failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO. ~~The minipurge valves are capable of closing under accident conditions. Therefore, they are allowed to be open for limited periods during power operation.~~ 2

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The purge valves must be maintained sealed closed ~~[or have blocks installed to prevent full opening]. [Blocked purge valves also actuate on an automatic signal.]~~ 3 The valves covered by this LCO are listed with their associated stroke times in the FSAR (Ref. 1). 5 U

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves or devices are those listed in Reference ~~2~~ 1. 2

BASES

LCO (continued)

Purge valves with resilient seals ~~[~~and secondary containment bypass valves~~]~~ must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing. 3

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

48

The ACTIONS are modified by a Note allowing penetration flow paths, except for ~~[42]~~ inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, these valves may not be opened under administrative controls. 3

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures that appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

BASES

ACTIONS (continued)

A fourth Note has been added that requires entry into the applicable Conditions and Required Actions of LCO 3.6.1 when leakage results in exceeding the overall containment leakage limit.

---------- REVIEWER'S NOTE ----------
The bracketed phrase "following isolation" in the Completion Times for Required Actions A.2, B.2, D.2, F.2 and F.3 is only applicable to plants that have adopted a Risk Informed Completion Time Program. 4

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time [or in accordance with the Risk Informed Completion Time Program]. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4 (Refs. 4 and 5). 2

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days [following isolation] for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the

BASES

ACTIONS (continued)

~~previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.~~

~~Condition A has been modified by a Note indicating that this Condition is only applicable to [the containment sump supply valves to the ECCS and containment spray pumps].~~

~~REVIEWER'S NOTE~~
~~Condition A is only applicable to the containment isolation valves that do not meet the conditions to extend the Completion Time to 7 days.~~

~~Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.~~

A

[~~B~~.1 and ~~B~~.2

~~REVIEWER'S NOTE~~
~~Adoption of the 7 day Completion Time is contingent on the conditions identified in Reference 4.~~

~~1. Individual licensees requesting CIV Completion Time relaxations should state in their plant-specific application that they have verified that the Joint Applications Report (JAR) results apply to their plant. Licensees should verify that the relaxed Completion Times will only apply to penetrations analyzed to meet the risk guidelines of Regulatory Guide 1.177 and fall within the 14 containment penetration configurations considered in the JAR. Any other containment penetration configurations not analyzed in the JAR must be supported by a plant-specific analysis. Licensee submittals must retain the current Completion Times for the three configurations that~~

BASES

ACTIONS (continued)

~~were not analyzed in the JAR: containment sump supply valves to the ECCS and containment spray systems pumps, valves associated with the main feedwater system, and main steam isolation valves.~~

~~2. Licensees should provide sufficient quantitative or qualitative substantiation to demonstrate that external events will not affect the results of the analysis supporting the extended Completion Times.~~

~~3. Licensees should state that they have verified acceptable PRA quality as described in Regulatory Guide 1.177.~~

~~4. Licensees should require verification of the operability of the remaining CIV(s) in a penetration flow path before entering the extended Completion Time for corrective maintenance. The JAR assumes that the penetrations remain physically intact in MODES in which these valves are to be operable during corrective maintenance. Licensees should describe in their plant specific application how the affected penetration will remain physically intact, or state that the penetration will be isolated so as to not permit a release to the outside environment.~~

~~5. The licensee should consider the additive nature of multiple failed CIVs, and the possibility of entering multiple AOTs and verify that these situations will result in risks consistent with the incremental conditional core damage probability (ICCDP) and incremental large early release probability (ICLERP) guidelines so that defense-in-depth for the safety systems will be maintained.~~

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for ~~Condition A and for~~ purge valve leakage and ~~shield building~~ bypass leakage not within limit~~]~~, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action ~~B~~.1, the device used to isolate the penetration should be the closest available one to containment. Required Action ~~B~~.1 must be completed within the ~~[7 day]~~ Completion Time ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The ~~[7 day]~~ Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4 ~~(References 3 and 4)~~.

BASES

ACTIONS (continued)

4 hour

For affected penetration flow paths that cannot be restored to OPERABLE status within the ~~[7 day]~~ Completion Time and that have been isolated in accordance with Required Action ~~B~~.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days ~~[~~following isolation~~]~~ for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

A 3 2

A

Condition ~~B~~ has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two ~~[or more]~~ containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition ~~D~~ provides appropriate actions.

2 3 2

C

A

Required Action ~~B~~.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.~~]~~

2 3

BASES

ACTIONS (continued)

B

~~C~~.1

With two ~~[or more]~~ containment isolation valves in one or more penetration flow paths inoperable, ~~[~~except for purge valve leakage and ~~shield building~~ bypass leakage not within limit~~]~~, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action ~~C~~.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action ~~B~~.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

secondary containment

B

A

B

Condition ~~C~~ is modified by a Note indicating this Condition is only applicable to penetration flow paths with two ~~[or more]~~ containment isolation valves. Condition ~~B~~ of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

A

C

~~D~~.1 and ~~D~~.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action ~~D~~.1 must be completed within the ~~[~~72~~]~~ hour Completion Time ~~for those penetrations that do not meet the 7 day Completion Time criteria and [7 days] for penetrations that do meet the 7 day Completion Time criteria~~. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ The specified time period is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during

C

BASES

ACTIONS (continued)

MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action ~~D~~.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days ~~[~~following isolation~~]~~ for verifying that each affected penetration flow path is isolated is appropriate considering the valves are operated under administrative controls and the probability of their misalignment is low.

Condition ~~D~~ is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference ~~4~~. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action ~~D~~.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

~~[E~~. 1

With the secondary containment bypass leakage rate (SR 3.6.3.~~9~~) ~~[~~or purge valve leakage rate (SR 3.6.3.~~6~~)~~]~~ not within limit, the assumptions of the safety analysis are not met. Therefore, the leakage must be restored to within limit. Restoration can be accomplished by isolating the penetration(s) that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When ~~a~~ penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time for secondary

BASES

ACTIONS (continued)

containment bypass leakage is reasonable considering the time required to restore the leakage by isolating the penetration(s) and the relative importance of secondary containment bypass leakage to the overall containment function. [The 24 hour Completion Time for purge valve leakage is acceptable considering the purge valves remain closed so that a gross breach of containment does not exist.]

to restore

REVIEWER'S NOTE

[The bracketed options provided in ACTION E reflect options in plant design and options in adopting the associated leakage rate Surveillances.

The options (in both ACTION E and ACTION F for purge valve leakage, are based primarily on the design - if leakage rates can be measured separately for each purge valve, ACTION F is intended to apply. This would be required to be able to implement Required Action F.3. Should the design allow only for leak testing both purge valves simultaneously, then the Completion Time for ACTION E should include the "24 hours for purge valve leakage" and ACTION F should be eliminated.]]

[F.1, F.2, and F.3

In the event one or more containment purge valves in one or more penetration flow paths are not within the purge valve leakage limits, purge valve leakage must be restored to within limits, or the affected penetration must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a [closed and de-activated automatic valve with resilient seals, a closed manual valve with resilient seals, or a blind flange]. A purge valve with resilient seals utilized to satisfy Required Action F.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.6. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action F.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification that those isolation devices outside

BASES

ACTIONS (continued)

~~containment capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.~~

~~For the containment purge valve with resilient seal that is isolated in accordance with Required Action F.1, SR 3.6.3.6 must be performed at least once every [92] days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.6, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 6). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per [92] days was chosen and has been shown to be acceptable based on operating experience.~~

~~Required Action F.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.]~~

E

~~G~~.1 and ~~G~~.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

BASES

ACTIONS (continued)

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~7~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action ~~G~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

~~[~~SR 3.6.3.1

This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a ~~[42]~~ inch containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A containment purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak

BASES

SURVEILLANCE REQUIREMENTS (continued)

tightness. ~~[The Frequency of 31 days is a result of an NRC initiative, Generic Issue B-24 (Ref. 8), related to containment purge valve use during unit operations.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

D

This SR is not required to be met while in Condition ~~E~~ of this LCO. This is reasonable since the penetration flow path would be isolated.~~]~~

, the time the leakage is not within limits is restricted, and if the purge valve leakage results in exceeding the overall containment leakage rate acceptance criteria, action will be required in accordance with LCO 3.6.1.

~~SR 3.6.3.2~~

~~This SR ensures that the minipurge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The minipurge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. [The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.~~3~~

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. ~~[Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

Containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, ~~4~~ and for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.~~4~~

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed,

BASES

SURVEILLANCE REQUIREMENTS (continued)

or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time that they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

4

SR 3.6.3.~~5~~ 2

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis.

~~----------REVIEWER'S NOTE----------~~
~~If the testing is within the scope of the licensee's INSERVICE TESTING PROGRAM, the Frequency "In accordance with the INSERVICE TESTING PROGRAM" should be used. Otherwise, the periodic Frequency of [18 months] or the reference to the Surveillance Frequency Control Program should be used.~~
~~----------~~ 4

~~[~~The Frequency of this SR is in accordance with ~~[~~the INSERVICE TESTING PROGRAM~~]~~ ~~[92 days.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.]]~~ 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 3

SR 3.6.3.~~6~~ 5 2

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option ~~[A]~~[B~~]~~, (Ref. ~~9~~), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. ~~[Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 6).~~ 4 2 2 3

Leakage testing of the purge valves in each penetration flow path is performed simultaneously and OPERABILITY is demonstrated by verifying that the measured leakage rate is ≤ 0.05 La when pressurized to Pa.

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 3

~~Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval is a prudent measure after a valve has been opened.~~ 2

SR 3.6.3.~~7~~ 6 2

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures each automatic containment isolation

BASES

SURVEILLANCE REQUIREMENTS (continued)

valve will actuate to its isolation position on a containment isolation actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~[The [18] month Frequency was developed considering it is prudent that this SR be performed only during a unit outage, since isolation of penetrations would eliminate cooling water flow and disrupt normal operation of many critical components. Operating experience has shown that these components usually pass this SR when performed on the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

~~[SR 3.6.3.8~~

~~---------- REVIEWER'S NOTE ----------~~
~~This SR is only required for those units with resilient seal purge valves allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices on the valves that are not permanently installed.~~
~~----------~~

~~Verifying that each [42] inch containment purge valve is blocked to restrict opening to ≤ [50]% is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of [recently] irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. [The [18] month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.~~ 3

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]]~~

~~[~~ SR 3.6. 3.~~9~~

This SR ensures that the combined leakage rate of all secondary containment bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. The Frequency is required by the Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria.

~~[~~Bypass leakage is considered part of L_a. unless specifically exempted.~~]]~~

REFERENCES

1. FSAR, Section ~~[ ]~~.

2. ~~FSAR, Section [ ].~~

3. ~~Combustion Engineering Owners Group (CEOG) Joint Applications Report (JAR) CE-NPSD-1168, Joint Applications Report for Containment Isolation Valve AOT Extension, dated June 1999.~~

4. ~~NRC Safety Evaluation for CEOG Joint Applications Report CE-NPSD-1168, "JAR for CIV AOT Extension," dated June 26, 2000.~~

~~5~~. Standard Review Plan 6.2.4.

6. ~~Generic Issue B-20.~~

~~7~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

BASES

REFERENCES (continued)

~~8. Generic Issue B-24.~~

4

~~9~~. 10 CFR 50, Appendix J, Option ~~[A][~~B~~]~~. 2 3

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves ~~(Atmospheric and Dual)~~ 1

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on an automatic isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. One of these barriers may be a closed system.

, Safety Injection Actuation Signal

containment high radiation

Containment isolation occurs upon receipt of a high containment pressure signal or a ~~low Reactor Coolant System (RCS) pressure~~ signal. The containment isolation signal closes automatic containment isolation valves in fluid penetrations not required for operation of Engineered Safety Feature systems in order to prevent leakage of radioactive material. Upon actuation of safety injection, automatic containment isolation valves also isolate systems not required for containment or RCS heat removal. Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated in the event of a release of radioactive material to containment atmosphere from the RCS following a Design Basis Accident (DBA). 2

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the accident analysis will be maintained.

8 inch

(48 inch)

8 inch supply

8 inch

The purge valves were designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor RCS leakage prior to personnel entry into containment. There are two sets of purge valves: normal purge and exhaust valves and ~~mini~~purge and exhaust valves. The normal and ~~mini~~purge supply and exhaust lines are each supplied with inside and outside containment isolation valves but share common supply and exhaust penetration lines. 2

BASES

BACKGROUND (continued)

Air operated isolation valves are provided both inside and outside the containment.

The normal purge valves are designed for purging the containment atmosphere to the unit stack while introducing filtered makeup from the outside to provide adequate ventilation for personnel comfort when the unit is shut down during refueling operations and maintenance. ~~Motor operated isolation valves are provided inside the containment, and air operated isolation valves are provided outside the containment.~~ The valves are operated manually from the control room. The valves will close automatically upon receipt of a containment ~~purge~~ isolation signal~~.~~ ~~The air operated valves~~ fail closed upon a loss of air. Because of their large size, the normal purge valves ~~in some units~~ are not qualified for automatic closure from their open position under DBA conditions. Therefore, the normal purge valves are normally maintained closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained. 2

and

8 inch

Open normal purge valves, or a failure of the ~~mini~~purge valves to close, following an accident that releases contamination to the containment atmosphere would cause a significant increase in the containment leakage rate. 2

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), a main steam line break, and a control element assembly ejection accident. In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analysis assumes that the normal purge valves are closed at event initiation.

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The single failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. ~~The inboard and outboard isolation valves on each line are provided with diverse power sources, motor operated and pneumatically operated spring closed, respectively.~~ This arrangement was designed to preclude common mode failures from disabling both valves on a purge line. 2

The purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valves due to failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO. The ~~mini~~purge valves are capable of closing under accident conditions. Therefore, they are allowed to be open for limited periods during power operation. 8 inch 2

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The purge valves must be maintained sealed closed ~~[or have blocks installed to prevent full opening]. [Blocked purge valves also actuate on an automatic signal.]~~ The valves covered by this LCO are listed with their associated stroke times in the FSAR (Ref. 1). 3 5 U

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves or devices are those listed in Reference ~~2~~. 2 1

BASES

LCO (continued)

Purge valves with resilient seals ~~[~~and secondary containment bypass valves~~]~~ must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing. 3

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

48

The ACTIONS are modified by a Note allowing penetration flow paths, except for ~~[42]~~ inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, these valves may not be opened under administrative controls. 3

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures that appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

BASES

ACTIONS (continued)

A fourth Note has been added that requires entry into the applicable Conditions and Required Actions of LCO 3.6.1 when leakage results in exceeding the overall containment leakage limit.

---------- REVIEWER'S NOTE ----------
The bracketed phrase "following isolation" in the Completion Times for Required Actions A.2, B.2, D.2, F.2 and F.3 is only applicable to plants that have adopted a Risk Informed Completion Time Program. 4

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time [or in accordance with the Risk Informed Completion Time Program]. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4 (Refs. 4 and 5). 2

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days [following isolation] for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the

BASES

ACTIONS (continued)

~~previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.~~

~~Condition A has been modified by a Note indicating that this Condition is only applicable to [the containment sump supply valves to the ECCS and containment spray pumps].~~

~~REVIEWER'S NOTE~~
~~Condition A is only applicable to the containment isolation valves that do not meet the conditions to extend the Completion Time to 7 days.~~

~~Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.~~

A

[~~B~~.1 and ~~B~~.2

~~REVIEWER'S NOTE~~
~~Adoption of the 7 day Completion Time is contingent on the conditions identified in Reference 4.~~

~~1. Individual licensees requesting CIV Completion Time relaxations should state in their plant specific application that they have verified that the Joint Applications Report (JAR) results apply to their plant. Licensees should verify that the relaxed Completion Times will only apply to penetrations analyzed to meet the risk guidelines of Regulatory Guide 1.177 and fall within the 14 containment penetration configurations considered in the JAR. Any other containment penetration configurations not analyzed in the JAR must be supported by a plant specific analysis. Licensee submittals must retain the current Completion Times for the three configurations that~~

BASES

ACTIONS (continued)

~~were not analyzed in the JAR: containment sump supply valves to the ECCS and containment spray systems pumps, valves associated with the main feedwater system, and main steam isolation valves.~~

~~2. Licensees should provide sufficient quantitative or qualitative substantiation to demonstrate that external events will not affect the results of the analysis supporting the extended Completion Times.~~

~~3. Licensees should state that they have verified acceptable PRA quality as described in Regulatory Guide 1.177.~~

~~4. Licensees should require verification of the operability of the remaining CIV(s) in a penetration flow path before entering the extended Completion Time for corrective maintenance. The JAR assumes that the penetrations remain physically intact in MODES in which these valves are to be operable during corrective maintenance. Licensees should describe in their plant specific application how the affected penetration will remain physically intact, or state that the penetration will be isolated so as to not permit a release to the outside environment.~~

~~5. The licensee should consider the additive nature of multiple failed CIVs, and the possibility of entering multiple AOTs and verify that these situations will result in risks consistent with the incremental conditional core damage probability (ICCDP) and incremental large early release probability (ICLERP) guidelines so that defense-in-depth for the safety systems will be maintained.~~

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for ~~Condition A and for~~ purge valve leakage and ~~shield building~~ bypass leakage not within limit~~]~~, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action ~~B~~.1, the device used to isolate the penetration should be the closest available one to containment. Required Action ~~B~~.1 must be completed within the ~~[7 day]~~ Completion Time ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The ~~[7 day]~~ Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4 ~~(References 3 and 4)~~.

BASES

ACTIONS (continued)

For affected penetration flow paths that cannot be restored to OPERABLE status within the ~~[7 day]~~ Completion Time and that have been isolated in accordance with Required Action ~~B~~.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days ~~[~~following isolation~~]~~ for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition ~~B~~ has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two ~~[or more]~~ containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition ~~D~~ provides appropriate actions.

Required Action ~~B~~.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.~~]~~

BASES

ACTIONS (continued)

B

~~C~~.1

With two ~~[or more]~~ containment isolation valves in one or more penetration flow paths inoperable, ~~[~~except for purge valve leakage and ~~shield building~~ bypass leakage not within limit~~]~~, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action ~~C~~.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action ~~B~~.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

secondary containment

B

A

B

Condition ~~C~~ is modified by a Note indicating this Condition is only applicable to penetration flow paths with two ~~[or more]~~ containment isolation valves. Condition ~~B~~ of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

A

C

~~D~~.1 and ~~D~~.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action ~~D~~.1 must be completed within the ~~[~~72~~]~~ hour Completion Time ~~for those penetrations that do not meet the 7 day Completion Time criteria and [7 days] for penetrations that do meet the 7 day Completion Time criteria~~. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ The specified time period is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during

C

BASES

ACTIONS (continued)

MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action ~~D~~.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days ~~[~~following isolation~~]~~ for verifying that each affected penetration flow path is isolated is appropriate considering the valves are operated under administrative controls and the probability of their misalignment is low.

Condition ~~D~~ is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference ~~4~~. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action ~~D~~.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

~~[~~ ~~E~~. 1

With the secondary containment bypass leakage rate (SR 3.6.3.~~9~~) ~~[or purge valve leakage rate (SR 3.6.3.6)]~~ not within limit, the assumptions of the safety analysis are not met. Therefore, the leakage must be restored to within limit. Restoration can be accomplished by isolating the penetration(s) that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time for secondary

BASES

ACTIONS (continued)

containment bypass leakage is reasonable considering the time required to restore the leakage by isolating the penetration(s) and the relative importance of secondary containment bypass leakage to the overall containment function. ~~[The 24 hour Completion Time for purge valve leakage is acceptable considering the purge valves remain closed so that a gross breach of containment does not exist.]~~

~~----------REVIEWER'S NOTE----------~~
~~[The bracketed options provided in ACTION E reflect options in plant design and options in adopting the associated leakage rate Surveillances.~~

~~The options (in both ACTION E and ACTION F for purge valve leakage, are based primarily on the design - if leakage rates can be measured separately for each purge valve, ACTION F is intended to apply. This would be required to be able to implement Required Action F.3. Should the design allow only for leak testing both purge valves simultaneously, then the Completion Time for ACTION E should include the "24 hours for purge valve leakage" and ACTION F should be eliminated.]]~~
~~----------~~

E

[~~F~~.1, ~~F~~.2, and ~~F~~.3

In the event one or more ~~containment purge valves in one or more~~ penetration flow paths are not within the purge valve leakage limits, purge valve leakage must be restored to within limits, or the affected penetration must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a ~~[~~closed and de-activated automatic valve with resilient seals, a closed manual valve with resilient seals, or a blind flange~~]~~. A ~~purge~~ valve with resilient seals utilized to satisfy Required Action ~~F~~.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.6. The specified Completion Time is reasonable, considering that one ~~containment purge~~ valve remains closed so that a gross breach of containment does not exist.

E

with a resilient seal

E

In accordance with Required Action ~~F~~.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification that those isolation devices outside

BASES

ACTIONS (continued)

containment capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the ~~containment purge~~ valve with resilient seal that is isolated in accordance with Required Action ~~F~~.1, SR 3.6.3.6 must be performed at least once every ~~[~~92~~]~~ days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the ~~containment purge valve~~ does not increase during the time the penetration is isolated. ~~The normal Frequency for SR 3.6.3.6, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 6).~~ Since more reliance is placed on a single ~~valve~~ while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per ~~[~~92~~]~~ days was chosen and has been shown to be acceptable based on operating experience.

Required Action ~~F~~.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. ~~]~~

~~G~~.1 and ~~G~~.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

BASES

ACTIONS (continued)

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~7~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action ~~G~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

~~[~~SR 3.6.3.1

This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a ~~[42]~~ inch containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A containment purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak

BASES

SURVEILLANCE REQUIREMENTS (continued)

tightness. ~~[The Frequency of 31 days is a result of an NRC initiative, Generic Issue B-24 (Ref. 8), related to containment purge valve use during unit operations.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

This SR is not required to be met while in Condition E of this LCO. This is reasonable since the penetration flow path would be isolated. ~~]~~

SR 3.6.3.2

8 inch

This SR ensures that the ~~mini~~purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The ~~mini~~purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. ~~[The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.~~

8 inch

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. ~~[Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

Containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 and for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed,

BASES

SURVEILLANCE REQUIREMENTS (continued)

or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time that they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis.

~~REVIEWER'S NOTE~~
~~If the testing is within the scope of the licensee's INSERVICE TESTING PROGRAM, the Frequency "In accordance with the INSERVICE TESTING PROGRAM" should be used. Otherwise, the periodic Frequency of [18 months] or the reference to the Surveillance Frequency Control Program should be used.~~

~~[~~The Frequency of this SR is in accordance with ~~[~~the INSERVICE TESTING PROGRAM~~]~~ ~~[92 days~~.

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.]]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 3

SR 3.6.3.6

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option ~~[A]~~[B~~]~~, (Ref. ~~9~~), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. ~~[~~ Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 6).~~

4

Leakage testing of the purge valves in each penetration flow path is performed simultaneously and OPERABILITY IS demonstrated by verifying that the measured leakage rate is ≤ 0.05 La when pressurized to Pa.

2 2 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 3

~~Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval is a prudent measure after a valve has been opened.~~ 2

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures each automatic containment isolation

BASES

SURVEILLANCE REQUIREMENTS (continued)

valve will actuate to its isolation position on a containment isolation actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~[The [18] month Frequency was developed considering it is prudent that this SR be performed only during a unit outage, since isolation of penetrations would eliminate cooling water flow and disrupt normal operation of many critical components. Operating experience has shown that these components usually pass this SR when performed on the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

~~[SR 3.6.3.8~~

~~---------- REVIEWER'S NOTE ----------~~
~~This SR is only required for those units with resilient seal purge valves allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices on the valves that are not permanently installed.~~
~~----------~~

~~Verifying that each [42] inch containment purge valve is blocked to restrict opening to ≤ [50]% is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of [recently] irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. [The [18] month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.~~ 3

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]]~~

~~[~~ SR 3.6.3.~~9~~

This SR ensures that the combined leakage rate of all secondary containment bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. The Frequency is required by the Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria.

~~[~~Bypass leakage is considered part of L_a. unless specifically exempted.~~]]~~

REFERENCES

1. FSAR, Section ~~[~~ ~~]~~.

2. ~~FSAR, Section [ ].~~

3. ~~Combustion Engineering Owners Group (CEOG) Joint Applications Report (JAR) CE-NPSD-1168, Joint Applications Report for Containment Isolation Valve AOT Extension, dated June 1999.~~

4. ~~NRC Safety Evaluation for CEOG Joint Applications Report CE-NPSD-1168, "JAR for CIV AOT Extension," dated June 26, 2000.~~

~~5~~. Standard Review Plan 6.2.4.

6. ~~Generic Issue B-20.~~

~~7~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

BASES

REFERENCES (continued)

~~8. Generic Issue B-24.~~

4

~~9~~. 10 CFR 50, Appendix J, Option ~~[A][~~B~~]~~. 2 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.3 BASES, CONTAINMENT ISOLATION VALVES

1. The type of Containment (Atmospheric and Dual) is deleted since it is unnecessary (only one Containment Specification is used in the St. Lucie Plant (PSL)ITS). This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ITS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. The PSL Safety Analysis Report (SAR) is an updated version of the original Final Safety Analysis Report. Therefore, the proper acronym is UFSAR and is changed to reflect the document.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.6.3, CONTAINMENT ISOLATION VALVES

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 4

ITS 3.6.4, Containment Pressure

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

CONTAINMENT SYSTEMS

CONTAINMENT

A02

~~INTERNAL~~ PRESSURE

LIMITING CONDITION FOR OPERATION

LCO 3.6.4 3.6.1.4 ~~Primary~~ containment ~~internal~~ pressure shall be ~~maintained between -0.490 and +0.5~~ psig.

A02

≥ -0.49 psig and ≤ +0.50

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTION:

ACTION A With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least ~~HOT STANDBY~~ within the next 6 hours and in

ACTION B ~~HOT SHUTDOWN~~ within the following 6 hours. LCO 3.0.4.a is not applicable when entering ~~HOT SHUTDOWN~~.

MODE 4

MODE 3

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1 4.6.1.4 The primary containment internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

CONTAINMENT

A02

~~INTERNAL~~ PRESSURE

LIMITING CONDITION FOR OPERATION

LCO 3.6.4 3.6.1.4 ~~Primary~~ containment ~~internal~~ pressure shall be ~~maintained between -0.420 and +0.400~~ psig.

A02

≥ -0.42 psig and ≤ +0.40

Applicability **APPLICABILITY:** MODES 1, 2, 3, and 4.

ACTION:

ACTION A With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least ~~HOT STANDBY~~ within the next 6 hours and in

ACTION B ~~HOT SHUTDOWN~~ within the following 6 hours. LCO 3.0.4.a is not applicable when entering ~~HOT SHUTDOWN~~.

MODE 4

MODE 3

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1 4.6.1.4 The primary containment internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

DISCUSSION OF CHANGES
ITS 3.6.4, CONTAINMENT PRESSURE

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.6.1.4 states, "Primary containment internal pressure shall be maintained between -0.490 and +0.5 psig" and Unit 2 CTS 3.6.1.4 states, in part, "Primary containment internal pressure shall be maintained between -0.420 and +0.400 psig." ITS 3.6.4 states "Containment pressure shall be ≥ -0.49 psig and ≤ +0.50 psig (Unit 1) ≥ -0.42 psig and ≤ +0.40 (Unit 2)." Additionally, the title for CTS 3.6.1.4 is "Internal Pressure." The title for ITS 3.6.4 is "Containment Pressure." This changes the CTS by changing the title and changing the presentation of the LCO statement.

This change is a wording preference that does not change the requirements for Containment Pressure. This change is designated as an administrative change and is acceptable because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

1

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure ~~(Atmospheric and Dual)~~

1

≥ -0.49 psig and ≤ +0.50 psig

3.6.1.4 LCO 3.6.4 Containment pressure shall be ~~[Dual: > 14.375 psia and < 27 inches water gauge] [or] [Atmospheric: ≥ -0.3 psig and ≤ +1.5 psig]~~.

3

APPLICABILITY APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.6.1.4	SR 3.6.4.1	Verify containment pressure is within limits.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure ~~(Atmospheric and Dual)~~ 1

3.6.1.4 LCO 3.6.4 Containment pressure shall be ~~[Dual: > 14.375 psia and < 27 inches water gauge] [or] [Atmospheric: ≥ -0.3 psig and ≤ +1.5 psig]~~ ≥ -0.42 psig and ≤ +0.40. 3

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.6.1.4	SR 3.6.4.1	Verify containment pressure is within limits.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.4, CONTAINMENT PRESSURE

1. The type of Containment (Atmospheric and Dual) is deleted since it is unnecessary. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ITS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4~~B~~ Containment Pressure ~~(Dual)~~ 1

BASES

BACKGROUND

The containment pressure is limited, during normal operation, to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or main steam line break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential, with respect to the ~~outside atmosphere~~ shield building annulus, in the event of inadvertent actuation of the Containment Spray System. 2

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered for determining the maximum containment internal pressure (P_a) are the LOCA and MSLB. Double-ended hot leg LOCA A~~n MSLB at 75% RTP~~ 2 results in the highest calculated internal containment pressure of 42.77 ~~42.3~~ psig, which is below the internal design pressure of 44 psig. The postulated DBAs are analyzed assuming degraded containment Engineered Safety Feature (ESF) systems (i.e., assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System and one train of the Containment Cooling System being rendered inoperable). It is this maximum containment pressure that is used to ensure that the licensing basis dose limitations are met.

The initial pressure condition used in the containment analysis was 15.51 psia ~~[14.7] psig~~. The maximum containment pressure resulting from the limiting DBA, 42.77 ~~[42.3]~~ psig, does not exceed the containment design pressure, ~~[~~44~~]~~ psig. The containment was also designed for an internal pressure equal to 1.04 ~~[0.65]~~ psid below external pressure to withstand the resultant pressure drop from an accidental actuation of the Containment Spray System. The LCO limit of -0.49 psig ~~[27] inches of water~~ ensures that operation within the design limit of -1.04 ~~[-0.65]~~ psid is maintained. The maximum calculated differential pressure that would occur as a result of an inadvertent actuation of the Containment Spray System between the containment and the shield building annulus is ~~[0.49]~~ psid. 3 (Ref. 1)

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). 0.9996 2

BASES

LCO

Maintaining containment pressure less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure greater than or equal to the LCO lower pressure limit ensures the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analysis are maintained, the LCO is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

~~---------- REVIEWER'S NOTE ----------~~
~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F]3, July 2000.~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~
~~---------- 4~~

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this

BASES

ACTIONS (continued)

status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~1~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 2 2

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4~~B~~.1 1

Verifying that containment pressure is within limits ensures that facility operation remains within the limits assumed in the containment analysis. ~~[The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

REFERENCES

~~1~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October 2001.

1. UFSAR Section 6.2

St. Lucie - Unit 1

Revision XXX

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4~~B~~ Containment Pressure ~~(Dual)~~

BASES

BACKGROUND

The containment pressure is limited, during normal operation, to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or main steam line break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential, with respect to the ~~outside atmosphere~~ shield building annulus, in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered for determining the maximum containment internal pressure (P_a) are the LOCA and MSLB. Double-ended hot leg LOCA ~~An MSLB at 75% RTP~~ results in the highest calculated internal containment pressure of 43.43 ~~42.3~~ psig, which is below the internal design pressure of 44 psig. The postulated DBAs are analyzed assuming degraded containment Engineered Safety Feature (ESF) systems (i.e., assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System and one train of the Containment Cooling System being rendered inoperable). It is this maximum containment pressure that is used to ensure that the licensing basis dose limitations are met.

The initial pressure condition used in the containment analysis was 15.41 psia ~~[14.7] psig~~. The maximum containment pressure resulting from the limiting DBA, ~~[42.3]~~ psig, does not exceed the containment design pressure, ~~[~~44~~]~~ psig. The containment was also designed for an internal pressure equal to 1.05 ~~[0.65]~~ psid below external pressure to withstand the resultant pressure drop from an accidental actuation of the Containment Spray System. The LCO limit of -0.42 psig ~~[27] inches of water~~ ensures that operation within the design limit of -1.05 ~~[-0.65]~~ psid is maintained. The maximum calculated differential pressure that would occur as a result of an inadvertent actuation of the Containment Spray System is ~~[0.49]~~ psid between the containment and the shield building annulus.

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). (Ref. 1) 0.9983

BASES

LCO

Maintaining containment pressure less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure greater than or equal to the LCO lower pressure limit ensures the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analysis are maintained, the LCO is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

~~---------- REVIEWER'S NOTE ----------~~
~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F]3, July 2000.~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

~~----------~~ 4

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this

BASES

ACTIONS (continued)

status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~1~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 2 2

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4~~B~~.1 1

Verifying that containment pressure is within limits ensures that facility operation remains within the limits assumed in the containment analysis. ~~[The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

REFERENCES

~~1~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October 2001.

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.4 BASES, CONTAINMENT PRESSURE

1. The type of Containment (Dual) and the Specification designator "B" are deleted since they are unnecessary (only one Containment Pressure Specification is used in the St. Lucie Plant (PSL) ITS.) This information is provided in NUREG-1432, Rev. 5.0, to assist in indentifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation. In addition, the Atmospheric Containment Pressure Bases (ISTS 3.6.4B) is not used and is not shown.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.6.4, CONTAINMENT PRESSURE

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

ITS 3.6.5, Containment Air Temperature

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

LCO 3.6.5

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

Applicability

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

ACTION A

ACTION B

Required Action B.2 NOTE

With the containment average air temperature > 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

4.6.1.5 ~~The primary~~ Verify containment average air temperature is within limit ~~shall be the arithmetical average of the temperatures at three of the following locations and shall be determined~~ in accordance with the Surveillance Frequency Control Program:

~~Location~~

~~a. Containment fan cooler No. 1A air intake, elevation 45 feet.~~

~~b. Containment fan cooler No. 1B air intake, elevation 45 feet.~~

~~c. Containment fan cooler No. 1C air intake, elevation 62 feet.~~

~~d. Containment fan cooler No. 1D air intake, elevation 45 feet.~~

LA01

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

LCO 3.6.5

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

ACTION A
ACTION B
Required Action B.2 NOTE

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

4.6.1.5 ~~The primary~~ Verify containment average air temperature is within limit ~~shall be the arithmetical average* of the temperatures at the following locations and shall be determined~~ in accordance with the Surveillance Frequency Control Program:

~~Location~~

~~a. TE-07-3A NW RCB Elevation 70'~~

~~b. TE-07-3B SW RCB Elevation 70'~~

LA01

~~* With one temperature detector inoperable, use the air intake temperature detectors of the operating containment fan coolers.~~

DISCUSSION OF CHANGES
ITS 3.6.5, CONTAINMENT AIR TEMPERATURE

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS) and additional Technical Specification Task Force (TSTF) travelers included in this submittal.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 4.6.1.5 requires the primary containment average air temperature to be the arithmetical average of the temperatures at three of four locations identified as containment fan cooler intakes No. 1A (Elevation 45 feet), No. 1B (Elevation 45 feet), No. 1C (Elevation 62 feet), No. 1D (Elevation 45 feet). Unit 2 CTS 4.6.1.5 requires the primary containment average air temperature to be the arithmetical average* of two temperature readings at Elevation 70'. Furthermore, Unit 2 CTS 4.6.1.5 requires the use of the air intake temperature detectors of the operating containment air coolers if one temperature detector is inoperable. R1

ITS SR 3.6.5.1 requires verification that the containment average air temperature is within the limit. This changes the CTS by moving the description of how compliance with the Technical Specification LCO is determined to the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to verify containment average air temperatures are within limits. Also, this change is acceptable because these types of procedural details will be adequately controlled in the Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural

details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

1

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature ~~(Atmospheric and Dual)~~

1

3.6.1.5 LCO 3.6.5 Containment average air temperature shall be ≤ ~~[~~120~~]~~°F.

3

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

2

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.6.1.5 SR 3.6.5.1	Verify containment average air temperature is within limit.	~~[~~ 24 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

CTS

1

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature ~~(Atmospheric and Dual)~~

1

3.6.1.5 LCO 3.6.5 Containment average air temperature shall be ≤ ~~[~~120~~]~~°F.

3

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

2

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.6.1.5 SR 3.6.5.1	Verify containment average air temperature is within limit.	~~[~~ 24 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.5, CONTAINMENT AIR TEMPERATURE

1. The type of Containment (Atmospheric and Dual) is deleted since it is unnecessary (only one Containment Specification is used in the St. Lucie Plant (PSL) ITS). This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature ~~(Atmospheric and Dual)~~

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or main steam line break (MSLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray and Cooling systems during post accident conditions is dependent on the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in a higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis (Ref. 1). Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analysis for containment. The accident analyses and evaluations considered both LOCAs and MSLBs for determining the maximum peak containment pressures and temperatures. The worst case MSLB generates larger mass and energy releases than the worst case LOCA. Thus, the MSLB event bounds the LOCA event from the containment peak ~~pressure and~~ temperature standpoint. The initial pre-accident temperature inside containment was assumed to be ~~[~~120~~]~~°F (Ref. ~~2~~).

~~[For atmospheric containment, the initial containment average air temperature condition of [120]°F resulted in a maximum vapor temperature in containment of [413]°F. The temperature of the~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~containment steel liner and concrete structure reach approximately 230°F and 220°F, respectively. The containment average air temperature limit of [120]°F ensures that, in the event of an accident, the maximum design temperature for containment, [300]°F, is not exceeded. The consequence of exceeding this design temperature may be the potential for degradation of the containment structure under accident loads.~~

~~For dual containment,~~ the initial containment condition of ~~[~~120~~]~~°F resulted in a maximum vapor temperature in containment of ~~[413.5]~~°F. ~~The~~ temperature of the containment steel pressure vessel ~~also reaches~~ approximately ~~[413.5]~~°F. The containment average temperature limit of ~~[~~120~~]~~°F ensures that, in the event of an accident, the maximum design temperature for containment of ~~[269.3]~~°F during LOCA ~~conditions~~ and ~~[413.5]°F during~~ MSLB conditions is not exceeded. The consequences of exceeding this design temperature may be the potential for degradation of the containment structure under accident loads.~~]~~

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature and that required safety related equipment will continue to perform its function.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

BASES

ACTIONS (continued)

B.1 and B.2

~~REVIEWER'S NOTE~~
~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F]3, July 2000.~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

4

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the 2 plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~3~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 2

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MOD 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere.

~~[The 24 hour Frequency of this SR is considered acceptable based on the observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

REFERENCES

1. FSAR, Section ~~[~~ ~~]~~.
2. ~~FSAR, Section [ ].~~
3. ~~3.~~ CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

2

INSERT 1

The arithmetical average is calculated from three of the following locations:

a. Containment fan cooler No. 1A air intake, elevation 45 feet,
b. Containment fan cooler No. 1B air intake, elevation 45 feet,
c. Containment fan cooler No. 1C air intake, elevation 62 feet, and
d. Containment fan cooler No. 1D air intake, elevation 45 feet.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature ~~(Atmospheric and Dual)~~ 1

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or main steam line break (MSLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray and Cooling systems during post accident conditions is dependent on the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in a higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis (Ref. 1). Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analysis for containment. The accident analyses and evaluations considered both LOCAs and MSLBs for determining the maximum peak containment pressures and temperatures. The worst case MSLB generates larger mass and energy releases than the worst case LOCA. Thus, the MSLB event bounds the LOCA event from the containment peak ~~pressure and~~ temperature standpoint. The initial pre-accident temperature inside containment was assumed to be ~~[~~120~~]~~°F (Ref. ~~2~~). 2 3 1 2

~~[For atmospheric containment, the initial containment average air temperature condition of [120]°F resulted in a maximum vapor temperature in containment of [413]°F. The temperature of the~~ 3

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~containment steel liner and concrete structure reach approximately 230°F and 220°F, respectively. The containment average air temperature limit of [120]°F ensures that, in the event of an accident, the maximum design temperature for containment, [300]°F, is not exceeded. The consequence of exceeding this design temperature may be the potential for degradation of the containment structure under accident loads.~~

~~For dual containment,~~ the initial containment condition of ~~[~~120~~]~~°F resulted in a maximum vapor temperature in containment of ~~[413.5]~~°F. ~~The~~ temperature of the containment steel pressure vessel ~~also reaches~~ approximately ~~[413.5]~~°F. The containment average temperature limit of ~~[~~120~~]~~°F ensures that, in the event of an accident, the maximum design temperature for containment of ~~[269.3]~~°F during LOCA ~~conditions~~ and ~~[413.5]°F during~~ MSLB conditions is not exceeded. The consequences of exceeding this design temperature may be the potential for degradation of the containment structure under accident loads.~~]~~

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature and that required safety related equipment will continue to perform its function.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

BASES

ACTIONS (continued)

B.1 and B.2

~~REVIEWER'S NOTE~~
~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F]3, July 2000.~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

4

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the 2 plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~3~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 2

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MOD 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere.

~~[The 24 hour Frequency of this SR is considered acceptable based on the observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

REFERENCES

1. FSAR, Section ~~[~~ 6.2 ~~]~~.
2. ~~FSAR, Section [ ].~~
3. ~~3.~~ CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

2

INSERT 1

The arithmetical average is calculated from the following containment temperature detectors:

a. TE-07-3A NW RCB Elevation 70', and
b. TE-07-3B SW RCB Elevation 70'.

When one of the two temperature detectors is not functional, the air intake temperature of the operating containment air coolers may be used.

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.5 BASES, CONTAINMENT AIR TEMPERATURE

1. The type of Containment (Atmospheric and Dual) is deleted since it is unnecessary (only one Containment Specification is used in the St. Lucie Plant (PSL) ITS). This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ITS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. The PSL Safety Analysis Report (SAR) is an updated version of the original Final Safety Analysis Report. Therefore, the proper acronym is UFSAR and is changed to reflect the document.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.6.5, CONTAINMENT AIR TEMPERATURE

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

ITS 3.6.6, Containment Spray and Cooling Systems

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

CONTAINMENT SYSTEMS

3/4.6.2 ~~DEPRESSURIZATION AND COOLING SYSTEMS~~

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

LCO 3.6.6

3.6.2.1 Two containment spray trains and two containment cooling trains shall be OPERABLE.

Applicability

APPLICABILITY: ~~Containment Spray System:~~ MODES 1, 2, ~~and MODE 3 with Pressurizer Pressure ≥ 1750 psia~~.

~~Containment Cooling System: MODES 1, 2, and 3~~.

and

Insert 1

ACTION:

1. Modes 1, 2, and 3 with Pressurizer Pressure ≥ 1750 psia:

L01

ACTION A

a. With one containment spray train inoperable, restore the inoperable spray train to OPERABLE status within ~~72 hours~~ or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within ~~the~~ ~~next~~ 6 hours and ~~in MODE 4~~ within ~~the following 54~~ hours.

7 days

12

ACTION F

M01

reduce pressurizer pressure to < 1750 psia

A02

ACTION B

b. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within ~~the next~~ 6 hours and ~~in MODE 4~~ within ~~the following 6~~ hours.

ACTION F

12

reduce pressurizer pressure to < 1750 psia

A02

ACTION D

c. ~~With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered inoperable.~~

Insert 2

L01

ACTION C NOTE

NOTE

Action not applicable when second containment spray train intentionally made inoperable.

LCO

10

ACTION C

d. With two containment spray trains inoperable, within 1 hour verify ~~TS~~ 3.7.~~7~~, "Control Room Emergency Ventilation System," is met, and restore at least one containment spray train to OPERABLE status within 24 hours; otherwise, be in MODE 3 within ~~the next~~ 6 hours and ~~in MODE 4 within the following 6~~ hours.

ACTION F

reduce pressurizer pressure to < 1750 psia within 12

A02

ACTION E

e. With two containment cooling trains inoperable, restore one cooling train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within ~~the~~ ~~next~~ 6 hours and ~~in MODE 4~~ within ~~the following 6~~ hours.

ACTION F

12

reduce pressurizer pressure to < 1750 psia

A02

ACTION I

f. With any combination of three or more trains inoperable, enter LCO 3.0.3 immediately

2. Mode 3 with Pressurizer Pressure < 1750 psia:

ACTION G

a. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours; otherwise be in MODE 4 within ~~the next 6~~ hours.

ACTION H

12

L02

ACTION I

b. With two containment cooling trains inoperable, enter LCO 3.0.3 immediately

A01

INSERT 1

MODE 3, except containment spray trains are not required to be OPERABLE when pressurizer pressure is < 1750.

L01

INSERT 2

D. One containment spray train and one containment cooling train inoperable in MODE 1, 2, or MODE 3 with pressurizer pressure ≥ 1750 psia.	D.1	Restore containment spray train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
	OR		
	D.2	Restore containment cooling train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

SR 3.6.6.1 a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is ~~positioned to take suction from the RWT on a Containment Pressure -- High High test signal~~.* LA01

In the correct position

SR 3.6.6.6 b. By verifying that each spray pump develops the specified discharge pressure when tested pursuant to the INSERVICE TESTING PROGRAM.

SR 3.6.6.5 c. In accordance with the Surveillance Frequency Control Program by verifying containment spray system locations susceptible to gas accumulation are sufficiently filled with water.

SR 3.6.6.1 NOTE * Not required to be met for system vent flow paths opened under administrative control.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

d. In accordance with the Surveillance Frequency Control Program~~, during shutdown,~~ by: L06

that is not locked, sealed, or otherwise secured in position, L03

SR 3.6.6.7 1. Verifying that each automatic valve in the flow path actuates to its correct position on ~~a CSAS test~~ signal.

an actual or simulated actuation

SR 3.6.6.8 2. Verifying that each spray pump starts automatically on ~~a CSAS test~~ signal. L04

an actual or simulated actuation

SR 3.6.6.4 3. Verify~~ing that upon a recirculation actuation signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established~~. LA02

Insert 3 L05

SR 3.6.6.10 e. By verifying each spray nozzle is unobstructed following maintenance which could result in nozzle blockage.

4.6.2.1.1. Each containment cooling train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

SR 3.6.6.2 1. Starting each cooling train fan unit from the control room and verifying that each unit operates for at least 15 minutes, and

SR 3.6.6.3 2. Verifying a cooling water flow rate of greater than or equal to 1200 gpm to each cooling unit.

SR 3.6.6.9 b. In accordance with the Surveillance Frequency Control Program~~, during shutdown,~~ by verifying that each containment cooling train starts automatically on an ~~SIAS test~~ signal. L06 L04

actual or simulated actuation

INSERT 3 L05 LA02

each automatic valve in the recirculation mode flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

LCO 3.6.6

3.6.2.1 Two containment spray trains and two containment cooling trains shall be OPERABLE.

Applicability

APPLICABILITY: ~~Containment Spray System:~~ MODES 1~~,~~ 2, ~~and MODE 3 with Pressurizer Pressure ≥ 1750 psia~~.

and

Insert 1

~~Containment Cooling System: MODES 1, 2, and 3~~.

ACTION:

1. Modes 1, 2, and 3 with Pressurizer Pressure ≥ 1750 psia:

ACTION A

a. With one containment spray train inoperable, restore the inoperable spray train to OPERABLE status within ~~72 hours~~ or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within ~~the~~ ~~next~~ 6 hours and ~~in MODE 4~~ within ~~the following 54~~ hours.

L01

7 days

ACTION F

12

M01

reduce pressurizer pressure to < 1750 psia

A02

ACTION B

b. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within ~~the next~~ 6 hours and ~~in MODE 4~~ within ~~the following 6~~ hours.

ACTION F

12

reduce pressurizer pressure to < 1750 psia

A02

ACTION D

c. ~~With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered inoperable.~~

Insert 2

L01

ACTION C NOTE

NOTE

Action not applicable when second containment spray train intentionally made inoperable.

Control Room Emergency Ventilation System

LCO

10

ACTION C

d. With two containment spray trains inoperable, within 1 hour verify 3.7.~~7~~, "~~CREACS~~," is met, and restore at least one containment spray train to OPERABLE status within 24 hours; otherwise, be in MODE 3 within the next 6 hours and ~~in MODE 4~~ within ~~the following 6~~ hours.

ACTION F

12

A02

reduce pressurizer pressure to < 1750 psia

ACTION E

e. With two containment cooling trains inoperable, restore one cooling train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within ~~the~~ ~~next~~ 6 hours and ~~in MODE 4~~ within ~~the following 6~~ hours.

ACTION F

12

A02

reduce pressurizer pressure to < 1750 psia

ACTION I

f. With any combination of three or more trains inoperable, enter LCO 3.0.3 immediately.

2. Mode 3 with Pressurizer Pressure < 1750 psia:

ACTION G

a. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours; otherwise be in MODE 4 within ~~the next 6~~ hours.

ACTION H

12

L02

ACTION I

b. With two containment cooling trains inoperable, enter LCO 3.0.3 immediately.

INSERT 1 A01

MODE 3, except containment spray trains are not required to be OPERABLE when pressurizer pressure is < 1750.

INSERT 2 L01

D. One containment spray train and one containment cooling train inoperable in MODE 1, 2, or MODE 3 with pressurizer pressure ≥ 1750 psia.	D.1	Restore containment spray train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
	OR		
	D.2	Restore containment cooling train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

SR 3.6.6.1

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is ~~positioned to take suction from the RWT on a Containment Pressure – – High-High test signal.~~* In the correct position LA01

SR 3.6.6.6

b. By verifying that each spray pump develops the specified discharge pressure when tested pursuant to the INSERVICE TESTING PROGRAM.

c. In accordance with the Surveillance Frequency Control Program~~, during shutdown,~~ L06 by:

that is not locked, sealed, or otherwise secured in position, L03

SR 3.6.6.7

1. Verifying that each automatic valve in the flow path actuates to its correct position on ~~a CSAS test~~ signal.

an actual or simulated actuation L04

SR 3.6.6.4

2. Verify~~ing that upon a Recirculation Actuation Test Signal (RAS), the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.~~ LA02

each automatic valve in the recirculation mode flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal. L05

SR 3.6.6.1 NOTE * Not required to be met for system vent flow paths opened under administrative control

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.6.6.8

3. Verifying that each spray pump starts automatically on ~~a CSAS test~~ signal. an actual or simulated actuation

L04

SR 3.6.6.5

d. In accordance with the Surveillance Frequency Control Program by verifying containment spray system locations susceptible to gas accumulation are sufficiently filled with water.

SR 3.6.6.10

e. By verifying each spray nozzle is unobstructed following maintenance which could result in nozzle blockage.

4.6.2.1.1. Each containment cooling train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

SR 3.6.6.2

1. Starting each cooling train fan unit from the control room and verifying that each unit operates for at least 15 minutes, and

SR 3.6.6.3

2. Verifying a cooling water flow rate of greater than or equal to 1200 gpm to each cooling unit.

SR 3.6.6.9

b. In accordance with the Surveillance Frequency Control Program~~, during shutdown,~~ by verifying that each containment cooling train starts automatically on an ~~SIAS test~~ signal. actual or simulated actuation

L06

L04

~~DELETED~~

~~DELETED~~

DISCUSSION OF CHANGES
ITS 3.6.6, CONTAINMENT SPRAY AND COOLING SYSTEMS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.6.2.1 Actions 1.a, 1.b, 1.d, and 1.e require in part, “otherwise be in… MODE 4...” when containment spray trains or cooling trains cannot be restored to OPERABLE status within the required time. ITS 3.6.6 Required Action F.2 requires, when the Required Actions and associated Completion Times are not met, reducing pressurizer pressure to < 1750 psia. This changes the end states in CTS 3.6.2.1 Action 1 from MODE 4 to MODE 3 with pressurizer pressure < 1750 psia.

The purpose of CTS 3.6.2.1 Action 1 is to place the plant in a condition in which the equipment is no longer required. CTS 3.6.2.1 Action 1 provides appropriate actions to take for the containment spray and containment cooling trains when the unit is in MODE 1, 2, or 3 when pressurizer pressure is ≥ 1750 psia. CTS 3.6.2.1 Action 2 provides appropriate actions to take for the containment cooling trains when the unit is in MODE 3 when pressurizer pressure is < 1750 psia. The Containment Spray System is not required to be OPERABLE when pressurizer pressure is < 1750 psia. Therefore, in accordance with CTS 3.0.2 (ITS LCO 3.0.2), when pressurizer pressure is < 1750 psia, the requirements to be in MODE 4 in CTS 3.6.2.1 Action 1 are no longer required to be completed. For the Containment Spray System, with pressurizer pressure below 1750 psia, the system is no longer required to be OPERABLE. For the Containment Cooling System, CTS 3.6.2.1 Action 2 applies and provides the appropriate actions while in MODE 3 with pressurizer pressure < 1750 psia.

These changes are designated as administrative changes and are acceptable because they maintain the current requirement to place the unit in a condition in which the applicable trains are no longer required consistent with the requirements of CTS 3.0.2 (ITS LCO 3.0.2) and, therefore, do not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.6.2.1 Action 1.a states in part, “otherwise be in… MODE 4 within the following 54 hours.” ITS 3.6.6 Required Action F.2, when the Required Actions and associated Completion Times are not met, requires reducing pressurizer pressure to < 1750 psia within 12 hours. This change reduces the time within which the plant must be placed in a condition where the containment spray trains are no longer required from 54 hours to 12 hours.

The purpose of CTS 3.6.2.1 Action 1.a is to place the plant in a condition in which the equipment is no longer required. The containment spray trains are only required to be OPERABLE when pressurizer pressure is ≥ 1750 psia. Therefore, in accordance with CTS 3.0.2 (ITS LCO 3.0.2), when pressurizer pressure is < 1750 psia, the requirements to be in MODE 4 in CTS 3.6.2.1 Action 1 are no longer required to be completed. This effectively allows 54 hours following entry into MODE 3 to reduce pressurizer pressure below 1750 psia. A Completion Time of 12 hours is a reasonable time to reach the required condition in an orderly manner without challenging plant systems. In addition, the 12 hour time is consistent with similar ITS Completion Times (e.g., Required Action C.2 of ISTS 3.5.1 and 3.5.2). The change is designated as more restrictive because it reduces a Required Action time to place the unit in a condition where the LCO does not apply.

RELOCATED SPECIFICATIONS

LA01 *(Type 2 – Removing Descriptions of System Operation)* CTS 4.6.2.1.a requires verification that each valve in the containment spray system is "positioned to take suction from the RWT on a Containment Pressure - - High-High test signal." ITS SR 3.6.6.1 requires verification that the valves are "in the correct position." This changes the CTS my moving the specific detail of system operation to the Bases.

The removal of these details, that are related to system operation, from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for the containment spray system to be demonstrated OPERABLE and the relocated material describes system operation and aspects of OPERABILITY. This change is acceptable because the removed information will be adequately controlled by the Technical Specification Bases Control Program in ITS Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system operation is being removed from the Technical Specification.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 4.6.2.1.d.3 and Unit 2 CTS 4.6.2.1.c.2 require "that upon a recirculation actuation signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established." ITS SR 3.6.6.4 requires verification that each automatic valve in the recirculation mode flow path actuates to the correct position on an actual or simulated actuation signal. This changes the CTS by moving the system detail of the recirculation mode alignment to the Bases. | R1

The removal of these details related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to verify that an automatic alignment of containment spray to the containment sump occurs on actual or simulated actuation signal. Also, this change is acceptable because the

removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specifications Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.6.2.1 ACTION 1.a for MODE 3 ≥ 1750 psia requires the restoration of a single inoperable containment spray train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program otherwise be in MODE 3 within 6 hours and MODE 4 within the following 54 hours. ITS 3.6.6 ACTION A requires restoration of the inoperable containment spray train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program otherwise ITS 3.6.6 ACTION F (discussed in DOCs A02 and M01) requires the plant to be in MODE 3 within 6 hours and pressurizer pressure reduced to <1750 psia within 12 hours. This changes the CTS by allowing 7 days or a period determined in accordance with the Risk Informed Completion Time Program to restore the inoperable containment spray train. | R1

This change is acceptable because the proposed time will still ensure the containment cooling safety function for Units 1 and 2 and the iodine removal safety function for Unit 1 are met. Unit 2 accident analyses does not credit containment spray for fission product removal. Each containment spray train is capable of satisfying at least 50% of the accident heat removal requirements and for Unit 1 100% of the post-accident iodine removal spray requirements. A redundant OPERABLE containment spray train in conjunction with two redundant 50% capacity containment cooling trains ensures a heat removal capability of at least 150% and iodine removal capability (Unit 1 only) of 100% for the 7 day duration. The 7 day period is consistent with the restoration time for an inoperable containment cooling train that is also capable of satisfying at least 50% of the accident heat removal requirements.

The 72 hour containment spray train restoration criterion is retained in ITS 3.6.6 ACTION D when the inoperable containment spray train occurs concurrent with an inoperable containment cooling train. In this condition, the capability of 100% accident containment heat and iodine removal (Unit 1 only) requirements are satisfied by the combination of one OPERABLE containment spray and one containment cooling train.

CTS ACTION 1.c. states, “With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered

inoperable." CTS ACTION 1.c. does not provide a specific time period requirement, instead relying on the times specified in CTS ACTION 1.a and CTS ACTION 1.b for the respective train inoperability. Because the inoperability of containment spray is the limiting component in CTS ACTION 1.c, (72 hours from CTS ACTION 1.a) restoration of the inoperable containment cooling train would necessarily have to be restored also within the 72 hours or less from the time the concurrent condition occurred to satisfy the CTS 1.c. ACTION requirement. The 7 days allowed in CTS ACTION 1.b is inconsequential if the available allowed time for restoration of an inoperable containment cooling train in accordance with CTS ACTION 1.b is (i.e., containment cooling train inoperable preceding the concurrent condition) $\geq$ 72 hours. If the available allowed time for restoration of the inoperable containment cooling train is < 72 hours the restoration time is limited by that remaining time. ITS ACTION D requires when one containment spray train and one containment cooling train are inoperable concurrently that one of the inoperable trains be restored to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program. Therefore, in effect there is no change to the CTS ACTION 1.c. restoration time requirement for an inoperable containment cooling train when the inoperability is concurrent with an inoperable containment spray train.

ITS ACTION F requires the plant be in MODE 3 in 6 hours and pressurizer pressure reduced to < 1750 psia in 12 hours if ITS 3.6.6 ACTIONS A, B, C, D, or E requirements are not met. This is consistent with the change to CTS discussed in DOC A03.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.6.2.1 ACTION 2.a for MODE 3 < 1750 psia requires, "With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours otherwise be in MODE 4 within the next 6 hours." ITS 3.6.6 ACTION H requires the plant to be in MODE 4 in 12 hours if an inoperable containment cooling train cannot be restored to OPERABLE status within 72 hours. This changes the CTS by allowing 12 hours to be in MODE 4.

This change is acceptable because based on operating experience a period of 12 hours to reach MODE 4 in an orderly manner and without challenging plant systems is reasonable and is consistent with similar Required Action Completion Times throughout the ITS (e.g., Required Action B.2 of ITS 3.6.4 and 3.6.5).

L03 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.6.2.1.d.1 and Unit 2 CTS 4.6.2.1.a.1 require verification that each automatic valve in the flow path actuates to its correct position on a CSAS test signal. ITS SR 3.6.6.7 requires verification that each automatic valve in the flow path actuates to the correct position except for valves that are locked, sealed, or otherwise secured in the actuated position. This changes the CTS by excluding

those valves that are locked, sealed, or otherwise secured in the actuated position from verification.

The purpose of Unit 1 CTS 4.6.2.1.d.1 and Unit 2 CTS 4.6.2.1.c.2 is to provide assurance that if an event occurred requiring actuation of the containment spray system, then those valves requiring automatic actuation would actuate to their correct position. Those automatic valves that are locked, sealed, or otherwise secured in position are not required to actuate on a containment spray actuation signal in order to perform their safety function because they are already in the required position. Testing such valves would not provide any additional assurance of OPERABILITY. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS. | R1

L04 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.6.2.1.d.1, CTS 4.6.2.1.d.2 and CTS 4.6.2.1.1.b and Unit 2 CTS 4.6.2.1.c.1, 4.6.2.1.c.3 and 4.6.2.1.1.b require verification that the containment spray system and containment cooling trains start automatically on a Containment Spray Actuation Signal (CSAS) test signal or Safety Injection Actuation Signal (SIAS), as applicable. ITS SR 3.6.6.7, ITS SR 3.6.6.8, and ITS SR 3.6.6.9 require verification that systems or components actuate on an actual or simulated actuation signal. This changes the CTS by explicitly allowing the use of either an actual or simulated signal for the tests. | R1

The purpose of Unit 1 CTS 4.6.2.1.d.1, CTS 4.6.2.1.d.2 and CTS 4.6.2.1.1.b and Unit 2 CTS 4.6.2.1.c.1, 4.6.2.1.c.3 and 4.6.2.1.1.b is to ensure the containment spray system and containment cooling trains start automatically on a Containment Spray Actuation Signal (CSAS) test signal or Safety Injection Actuation Signal (SIAS), as applicable. This change is acceptable because it has been determined that the current Surveillance Requirement acceptance criteria are not the only method that can be used for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual" or "simulated" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L05 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.6.2.1.d.3 and Unit 2 CTS 4.6.2.1.c.2 require verification that the containment spray system automatically aligns to the containment sump upon receipt of an RAS actuation signal. ITS SR 3.6.6.4 requires verification that each automatic valve in the recirculation mode flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal. This changes the CTS by explicitly allowing the use of either an actual or simulated signal for the tests.

The purpose of Unit 1 CTS 4.6.2.1.d.3 and Unit 2 CTS 4.6.2.1.c.2 is to ensure the containment spray system actuates to align the containment spray system to

the containment sump upon receipt of an RAS signal. This change is acceptable because it has been determined that the current Surveillance Requirement acceptance criteria are not the only method that can be used for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual" or "simulated" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L06 *(Category 8 – Deletion of Surveillance Requirement Shutdown Performance Requirements)* Unit 1 CTS 4.6.2.1.d, Unit 1 CTS 4.6.2.1.1.b, Unit 2 CTS 4.6.2.1.c and Unit 2 CTS 4.6.2.1.1.b state, in part, “In accordance with the Surveillance Frequency Control Program, during shutdown.” ITS surveillances state “In accordance with the Surveillance Frequency Control Program”. This changes the CTS by deleting the SR requirement to perform the Surveillances during shutdown.

This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The proposed Surveillance does not include the restriction on unit conditions. The Surveillance Requirements could be performed in other than shutdown conditions, without jeopardizing safe plant operations. The control of the unit conditions appropriate to perform the test is an issue for procedures and scheduling and has been determined by the NRC Staff to be unnecessary as a Technical Specification restriction. As indicated in Generic Letter 91-04, allowing this control is consistent with the vast majority of other Technical Specification Surveillances that do not dictate unit conditions for the Surveillance. This change is designated as less restrictive because the Surveillance may be performed at plant conditions other than shutdown.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.6 CONTAINMENT SYSTEMS

3.6.6~~A~~ Containment Spray and Cooling Systems ~~(Atmospheric and Dual)~~ ~~(Credit taken for iodine removal by the Containment Spray System)~~

3.6.2.1 LCO 3.6.6~~A~~ Two containment spray trains and two containment cooling trains shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2~~, 3, and [4].~~

and

MODE 3, except containment spray trains are not required to be OPERABLE when pressurizer pressure is < 1750 psia.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION 1.a. DOC L01 A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	~~[~~7~~]~~ days ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
ACTION 1.b. B. One containment cooling train inoperable. in MODE 1, 2, or MODE 3 with pressurizer pressure ≥ 1750 psia	B.1 Restore containment cooling train to OPERABLE status.	7 days ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
ACTION 1.d. NOTE C.---------- NOTE --------- Not applicable when second containment spray train intentionally made inoperable. ---------- ACTION 1.d. Two containment spray trains inoperable.	C.1 Verify LCO 3.7.1~~1~~0, "~~CREACS~~," is met. Control Room Emergency Ventilation System (CREVS) AND C.2 Restore at least one containment spray train to OPERABLE status.	1 hour 24 hours

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One containment spray train and one containment cooling train inoperable. (ACTION 1.c. DOC L01) (in MODE 1, 2, or MODE 3 with pressurizer pressure ≥ 1750 psia)	D.1 Restore containment spray train to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
	OR	
	D.2 Restore containment cooling train to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
E. Two containment cooling trains inoperable. (ACTION 1.e.) (in MODE 1, 2, or MODE 3 with pressurizer pressure ≥ 1750 psia)	E.1 Restore one containment cooling train to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
F. Required Action and associated Completion Time not met. (ACTION 1.a ACTION 1.b. ACTION 1.c. ACTION 1.d ACTION 1.e DOC A02 DOC M01) (of Condition A, B, C, D, or E)	F.1 Be in MODE 3. AND	6 hours
	F.2 ~~----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------~~ ~~Be in MODE 4~~. (Reduce pressurizer pressure to < 1750 psia)	12 hours

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION 1.f. ~~C~~. Any combination of three or more trains inoperable. in MODE 1, 2, or MODE 3 with pressurizer pressure ≥ 1750 psia Insert 2	~~C~~.1 Enter LCO 3.0.3.	Immediately Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6~~A~~.1	----------NOTE---------- Not required to be met for system vent flow paths opened under administrative control. ----------	
4.6.2.1.a	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 3.6.6~~A~~.2	Operate each containment cooling train fan unit for ≥ 15 minutes.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

4.6.2.1.a footnote

4.6.2.1.1.a.1

CTS

6

INSERT 1

ACTION 2.a.	G. One containment cooling train inoperable in MODE 3 with pressurizer pressure < 1750 psia.	G.1	Restore containment cooling train to OPERABLE status.	72 hours
ACTION 2.a DOC L02	H. Required Action and associated Completion Time of Condition G not met.	H.1	Be in MODE 4.	12 hours

2

INSERT 2

OR

ACTION 2.b.

Two containment cooling trains inoperable in MODE 3 with pressurizer pressure < 1750 psia.

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.6~~A~~.3	Verify each containment cooling train cooling water flow rate is ≥ ~~[2000]~~ gpm to each fan cooler. (1200)	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
SR 3.6.6~~A~~.4	~~[Verify the containment spray piping is full of water to the [100] ft level in the containment spray header.~~ Verify each automatic valve in the recirculation mode flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~
SR 3.6.6~~A~~.5	Verify containment spray locations susceptible to gas accumulation are sufficiently filled with water.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
SR 3.6.6~~A~~.6	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM

4.6.2.1.1.a.2
4.6.2.1.d.3
DOC L05
4.6.2.1.c
4.6.2.1.b

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6~~A~~.7	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 3.6.6~~A~~.8	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 3.6.6~~A~~.9	Verify each containment cooling train starts automatically on an actual or simulated actuation signal.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

4.6.2.1.d.1
DOC L03
DOC L04

4.6.2.1.d.2
DOC L04

4.6.2.1.1.b
DOC L04

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
4.6.2.1.d SR 3.6.6~~A~~.10	Verify each spray nozzle is unobstructed.	~~[At first refueling]~~ ~~AND~~ ~~[10 years~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

Following maintenance which could result in nozzle blockage

CTS

3.6 CONTAINMENT SYSTEMS

3.6.6~~A~~ Containment Spray and Cooling Systems ~~(Atmospheric and Dual)~~ ~~(Credit taken for iodine removal by the Containment Spray System)~~

3.6.2.1 LCO 3.6.6~~A~~ Two containment spray trains and two containment cooling trains shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2~~, 3, and [4].~~

and

MODE 3, except containment spray trains are not required to be OPERABLE when pressurizer pressure is < 1750 psia.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION 1.a. DOC L01 A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	~~[~~7~~]~~ days ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
ACTION 1.b. B. One containment cooling train inoperable. in MODE 1, 2, or MODE 3 with pressurizer pressure ≥ 1750 psia	B.1 Restore containment cooling train to OPERABLE status.	7 days ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
ACTION 1.d. NOTE C.---------- NOTE --------- Not applicable when second containment spray train intentionally made inoperable. ---------- ACTION 1.d. Two containment spray trains inoperable.	C.1 Verify LCO 3.7.1~~1~~0, "~~CREACS~~," is met. Control Room Emergency Ventilation System (CREVS) AND C.2 Restore at least one containment spray train to OPERABLE status.	1 hour 24 hours

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One containment spray train and one containment cooling train inoperable. (ACTION 1.c. DOC L01) (in MODE 1, 2, or MODE 3 with pressurizer pressure ≥ 1750 psia)	D.1 Restore containment spray train to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
	OR	
	D.2 Restore containment cooling train to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
E. Two containment cooling trains inoperable. (ACTION 1.e.) (in MODE 1, 2, or MODE 3 with pressurizer pressure ≥ 1750 psia)	E.1 Restore one containment cooling train to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
F. Required Action and associated Completion Time not met. (ACTION 1.a ACTION 1.b. ACTION 1.c. ACTION 1.d ACTION 1.e DOC A02 DOC M01) (of Condition A, B, C, D, or E)	F.1 Be in MODE 3. AND	6 hours
	F.2 ~~----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------~~ ~~Be in MODE 4~~. (Reduce pressurizer pressure to < 1750 psia)	12 hours

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION 1.f. ~~C~~. Any combination of three or more trains inoperable. in MODE 1, 2, or MODE 3 with pressurizer pressure ≥ 1750 psia	~~C~~.1 Enter LCO 3.0.3.	Immediately

Insert 1

Insert 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6~~A~~.1	----------NOTE---------- Not required to be met for system vent flow paths opened under administrative control. ----------	
4.6.2.1.a	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 3.6.6~~A~~.2	Operate each containment cooling train fan unit for ≥ 15 minutes.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

4.6.2.1.a footnote

4.6.2.1.1.a.1

CTS

6

INSERT 1

ACTION 2.a.	G. One containment cooling train inoperable in MODE 3 with pressurizer pressure < 1750 psia.	G.1	Restore containment cooling train to OPERABLE status.	72 hours
ACTION 2.a DOC L02	H. Required Action and associated Completion Time of Condition G not met.	H.1	Be in MODE 4.	12 hours

2

INSERT 2

OR

ACTION 2.b.

Two containment cooling trains inoperable in MODE 3 with pressurizer pressure < 1750 psia.

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.6~~A~~.3 4.6.2.1.1.a.2	Verify each containment cooling train cooling water flow rate is ≥ ~~[2000]~~ gpm to each fan cooler. 1200	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 3.6.6~~A~~.4 4.6.2.1.c.2 DOC L05	~~[Verify the containment spray piping is full of water to the [100] ft level in the containment spray header.~~ Verify each automatic valve in the recirculation mode flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~
SR 3.6.6~~A~~.5 4.6.2.1.d	Verify containment spray locations susceptible to gas accumulation are sufficiently filled with water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 3.6.6~~A~~.6 4.6.2.1.b	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.6.2.1.c.1 DOC L03 DOC L04	SR 3.6.6~~A~~.7	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
4.6.2.1.c.3 DOC L04	SR 3.6.6~~A~~.8	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
4.6.2.1.1.b DOC L04	SR 3.6.6~~A~~.9	Verify each containment cooling train starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
4.6.2.1.e SR 3.6.6~~A~~.10	Verify each spray nozzle is unobstructed.	~~[At first refueling]~~ ~~AND~~ ~~[10 years~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

Following maintenance which could result in nozzle blockage

St. Lucie Unit 2

Amendment XXX

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.6, CONTAINMENT SPRAY AND COOLING SYSTEMS

1. The type of Containment (Atmospheric and Dual), Specification designator "A" and the descriptor (credit taken for iodine removal by the Containment Spray System) are deleted since they are unnecessary (only one Containment Specification type is used in the St. Lucie Plant (PSL) Unit 1 and 2 ITS). ISTS 3.6.6A is used for PSL Unit 1 and Unit 2. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ITS which reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. The LCO 3.0.4.a Note to ISTS 3.6.6 Required Action F.2 is not applicable to the PSL ITS because the ITS 3.6.6 Applicability does not include MODE 4 rendering the Note unnecessary. Therefore, the Note has been removed.

5. The Control Room Emergency Air Cleanup System (ISTS 3.7.11) is renumbered and relabeled as ITS 3.7.10, "Control Room Emergency Ventilation System (CREVS)," because the PSL ITS does not include ISTS 3.7.10, "Essential Chilled Water (ECW) System.

6. PSL Current Technical Specifications only require the Containment Spray System to be OPERABLE in MODES 1, 2 and 3 when pressurizer pressure is ≥ 1750 psia. The Applicability of ISTS 3.6.6 reflects this current licensing basis. In addition, the actions associated with containment cooling trains are different based on pressurizer pressure above or below 1750 psia. The Condition qualifiers added to ISTS 3.6.6, Conditions B, D, E, F, and G (ITS Condition I) reflect Conditions that apply when pressurizer pressure is ≥ 1750 psia and the new ACTIONS provided in Insert 1 reflect Conditions that apply when pressurizer pressure is < 1750 psia.

   The change made to Required Action F.2 reflects the end state for Conditions that apply when in MODE 1, 2, or MODE 3 with pressurizer pressure is ≥ 1750 psia. The new ITS 3.6.6 Condition I reflects the end state for Conditions that apply when in MODE 3 with pressurizer pressure is < 1750 psia. Conditions and Required Actions are renumerated to reflect the added ITS ACTIONS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6~~A~~ Containment Spray and Cooling Systems ~~(Atmospheric and Dual)~~
~~(Credit taken for iodine removal by the Containment Spray System)~~

BASES

BACKGROUND

The Containment Spray and Containment Cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling systems are designed to the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1), or other documents that were appropriate at the time of licensing (identified on a unit specific basis).

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and the Containment Cooling System provide redundant methods to limit and maintain post accident conditions to less than the containment design values.

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray header~~s~~, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water tank (RWT) supplies borated water to the containment spray during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWT to the containment sump(s).

shutdown heat exchanger,

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide from the spray additive tank into the upper regions of containment to reduce containment pressure and temperature and to reduce the concentration of fission products in the containment atmosphere during a DBA. The RWT solution temperature is an

BASES

BACKGROUND (continued)

important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the shutdown cooling heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet 50% of the system design requirements for containment heat removal and 100% of the iodine removal design bases.

The Spray Additive System injects a ~~hydrazine (N_2H_4)~~ sodium hydroxide (NaOH) solution into the spray. The resulting alkaline pH of the spray enhances its ability to scavenge fission products from the containment atmosphere. The ~~N_2H_4~~ NaOH added to the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically by a containment High-High pressure signal coincident with a safety injection actuation signal (SIAS) or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two Containment Spray System pumps, and begins the injection phase. The containment spray header isolation valves open upon a containment spray actuation signal. A manual actuation of the Containment Spray System is available on the main control board to begin the same sequence. The injection phase continues until an RWT level Low signal is received. The Low level for the RWT generates a recirculation actuation signal that aligns valves from the containment spray pump suction to the containment sump. The Containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

Containment Cooling System

Two trains of containment cooling, each of sufficient capacity to supply 50% of the design cooling requirement, are provided. Two trains with two fan units each are supplied with cooling water from ~~a separate train of service water cooling~~ redundant essential cooling headers through normally open valves. All four fans are required to furnish the design cooling capacity. ~~Air is drawn into the coolers through the fans and discharged to the steam generator compartments and pressurizer compartment.~~

BASES

BACKGROUND (continued)

During normal operation, all four containment fan coolers are in operation. If a fan were to be idle, it would automatically be started on receipt of an SIAS.

~~In post accident operation following a containment cooling actuation signal (CCAS), all four Containment Cooling System fans are designed to start automatically in slow speed. Cooling is shifted from the chilled water cooled coils to the service water cooled coils.~~ The temperature of the ~~service~~ component cooling water and intake cooling water is an important factor in the heat removal capability of the fan units.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the main steam line break (MSLB). The DBA LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System and one train of the Containment Cooling System being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.77 ~~[55.7]~~ psig (experienced during a LOCA ~~n MSLB~~). The analysis shows that the peak containment vapor temperature is 387 ~~[413]~~ °F (experienced during an MSLB). Both results are within the design. (See the Bases for Specifications 3.6.4~~A and 3.6.4B~~, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of 100.3 ~~[102]~~% RTP, one containment spray train and one containment cooling train operating, and initial (pre-accident) conditions of ~~[~~120~~]~~°F and 15.51 ~~[14.7]~~ psia. The analyses also assume a response time delayed initiation in order to provide a conservative calculation of peak containment pressure and temperature responses.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in an annulus to ~~reduces the~~ containment pressure of 0.9996 psid ~~to [-2.8] psig~~ differential due to the sudden cooling effect in the interior of the air tight containment. Additional discussion is provided in the Bases for Specifications 3.6.4~~A, 3.6.4B~~, and 3.6.~~12~~ 8, "Vacuum Relief Valves."

The modeled Containment Spray System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint coincident with an SIAS to achieve full flow through the containment spray nozzles. The

BASES

APPLICABLE SAFETY ANALYSES (continued)

Containment Spray System total response time of ~~[60]~~ seconds includes diesel generator startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 2).

The performance of the containment cooling train for post accident conditions is given in Reference ~~3~~. The result of the analysis is that each train can provide 50% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference ~~4~~.

The modeled Containment Cooling System actuation from the containment analysis is based upon the unit specific response time associated with exceeding the ~~CCAS~~ to achieve full Containment Cooling System air and safety grade cooling water flow.

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, a minimum of two containment cooling trains or two containment spray trains, or one of each, is required to maintain the containment peak pressure and temperature below the design limits (Ref. ~~5~~). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming that the worst case single active failure occurs.

Each Containment Spray System includes a spray pump, spray header~~s~~, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT upon an ESF actuation signal and automatically transferring suction to the containment sump. Management of gas voids is important to Containment Spray System OPERABILITY.

Each Containment Cooling System includes ~~demisters,~~ cooling coils, ~~dampers,~~ fans, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, ~~3,~~ and ~~4~~, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the containment spray trains and containment cooling trains.

Containment spray trains are not required to be OPERABLE when pressurizer pressure is < 1750 psia.

BASES

APPLICABILITY (continued)

In MODES ~~5~~ and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the ~~Containment Spray and~~ Containment Cooling ~~systems are~~ not required to be OPERABLE in MODES ~~5~~ and 6.

ACTIONS

A.1

~~REVIEWER'S NOTE~~

~~Utilization of the 7 day Completion Time for Required Action A.1 is dependent on the licensee adopting CE NPSD-1045-A (Ref. 6) and meeting the requirements of the Topical Report and the associated Safety Evaluation including the following commitment: "[LICENSEE] has enhanced its Configuration Risk Management Program, as implemented under 10 CFR 50.65(a)(4), the Maintenance Rule, to include a Large Early Release Fraction assessment to support this application." Otherwise, a 72 hour Completion Time applies.~~

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within ~~[~~7~~]~~ days ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The ~~[~~7~~]~~ day Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and the findings of Ref. ~~6~~.

B.1

With one required containment cooling train inoperable, the inoperable containment cooling train must be restored to OPERABLE status within 7 days ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The components in this degraded condition are capable of providing greater than 100% of the heat removal needs (for the condition of one containment cooling train inoperable) after an accident. The 7 day Completion Time was developed based on the same reasons as those for Required Action A.1.

BASES

ACTIONS (continued)

C.1

and pressurizer pressure ≥ 1750 psia

With two required containment spray trains inoperable~~,~~ at least one of the required containment spray trains must be restored to OPERABLE status within 24 hours. Both trains of containment cooling must be OPERABLE or Condition ~~C~~ is also entered. The Condition is modified by a Note stating it is not applicable if the second containment spray train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. In addition, LCO 3.7.~~11~~, "~~CREACS~~," must be verified to be met within 1 hour. The components in this degraded condition are capable of providing greater than 100% of the heat removal needs after an accident. The Completion Time is based on Reference ~~7~~ which demonstrated that the 24 hour Completion Time is acceptable based on the redundant heat removal capabilities afforded by the Containment Cooling System, the iodine removal capability of ~~the Control Room Emergency Air Cleanup System~~, the infrequent use of the Required Action, and the small incremental effect on plant risk.

D.1 and D.2

with pressurizer pressure ≥ 1750 psia

With one containment spray train and one containment cooling train inoperable~~,~~ one of the required containment spray or containment cooling trains must be restored to OPERABLE status within 72 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System, the iodine removal function of the Containment Spray System, and the low probability of a DBA occurring during this period.

E.1

and pressurizer pressure ≥ 1750 psia

With two required containment cooling trains inoperable~~,~~ one of the required containment cooling trains must be restored to OPERABLE status within 72 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour

BASES

ACTIONS (continued)

Completion Time was developed taking into account the redundant heat removal capabilities afforded by ~~combinations of~~ the Containment Spray System ~~and Containment Cooling System~~, the iodine removal function of the Containment Spray System, and the low probability of a DBA occurring during this period.

F.1 and F.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

condition where the requirements do not apply. Containment spray trains are not required to be OPERABLE when pressurizer pressure is below 1750 psia and other Conditions apply when containment cooling trains are inoperable with pressurizer pressure below 1750 psia.

of Condition A, B, C, D, or E

If the Required Actions and associated Completion Times are not met, the plant must be brought to a ~~MODE in which overall plant risk is minimized~~. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and ~~to MODE 4~~ within 12 hours.

reduce pressurizer pressure to < 1750 psia

~~Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.~~

If one or more containment cooling trains remain inoperable once pressurizer pressure is < 1750 psia, the Conditions (i.e., Conditions G and I) must be entered, as applicable, and Required Actions performed within the associated Completion Times.

~~Required Action F.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the~~

BASES

ACTIONS (continued)

~~results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Insert 1

I

~~C~~.1

with pressurizer pressure ≥ 1750 psia

With any combination of three or more Containment Spray System and Containment Cooling System trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

Insert 2

SURVEILLANCE REQUIREMENTS

SR 3.6.6~~A~~.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This SR does not require any testing or valve manipulation. Rather, it involves verifying that those valves outside containment and capable of potentially being mispositioned are in the correct position.

~~[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

7

INSERT 1

G.1

With one required containment cooling train inoperable in MODE 3 with pressurizer pressure < 1750 psia, the inoperable containment cooling train must be restored to OPERABLE status within 72 hours. The remaining OPERABLE containment cooling train can provide sufficient heat removal needs after an accident when in MODE 3 and pressurizer pressure < 1750 psia. The 72 hours Completion Time was developed considering the reduced heat removal needs and the low probability of a DBA occurring during this period.

H.1

If the Required Action and associated Completion Time of Condition G are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 4 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant condition in an orderly manner.

7

INSERT 2

Additionally, with two containment cooling trains inoperable with pressurizer pressure < 1750 psia the containment cooling function may be lost since the Containment Spray System is not required to be OPERABLE below 1750 psia.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent flow path who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the system vent flow path if directed.

SR 3.6.6~~A~~.2

Operating each containment cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected and corrective action taken. ~~[The 31 day Frequency of this SR was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of a significant degradation of the containment cooling train occurring between surveillances and has been shown to be acceptable through operating experience.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6~~A~~.3

cooling

1200

Verifying a ~~service~~ water flow rate of ≥ ~~[2000]~~ gpm to each cooling unit provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 2). ~~[Also considered in selecting the Frequency of 31 days were the known reliability of the Cooling Water System, the two train redundancy, and the low probability of a significant degradation of flow occurring between surveillances.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~[~~ SR 3.6.6~~A~~.4

~~Verifying that the containment spray header piping is full of water to the [100] ft level minimizes the time required to fill the header. This ensures that spray flow will be admitted to the containment atmosphere within the time frame assumed in the containment analysis. [The 31 day Frequency is based on the static nature of the fill header and the low probability of a significant degradation of water level in the piping occurring between surveillances.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]]~~

This surveillance verifies that each automatic valve in the recirculation mode flow path actuates to the correct position on an actual or simulated actuation signal. The surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6~~A~~.5 1

Containment Spray System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the required containment spray trains and may also prevent water hammer and pump cavitation.

Selection of Containment Spray System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The Containment Spray System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the Containment Spray System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits. | R1

Containment Spray System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void

BASES

SURVEILLANCE REQUIREMENTS (continued)

volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the Containment Spray System piping and the procedural controls governing system operation.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.6.6~~A~~.6

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by the ASME Code (Ref. ~~9~~). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.6.6~~A~~.7 and SR 3.6.6~~A~~.8

These SRs verify that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~[The [18] month~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

The surveillance of containment sump isolation valves is also required by SR 3.5.2.~~5~~. A single surveillance may be used to satisfy both requirements.

SR 3.6.6~~A~~.9

This SR verifies that each containment cooling train actuates upon receipt of an actual or simulated actuation signal. ~~[The [18] month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6A.7 and SR 3.6.6A.8, above, for further discussion of the basis for the [18] month Frequency.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6~~A~~.10

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. Performance of this SR demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. ~~[~~Due to the passive design of the nozzle, a test ~~at [the first refueling and at] 10 year intervals~~ is considered adequate to detect obstruction of the spray nozzles following maintenance that could result in nozzle blockage.

Foreign material introduced as a result of maintenance, although unlikely, is the most likely potential cause for future spray nozzle obstruction. Therefore, verification conducted to confirm that potentially affected nozzles are free of blockage following maintenance activities that could result in nozzle blockage, is sufficient to confirm that the nozzles are free of obstruction.

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. FSAR, Section ~~[ ]~~.
3. ~~FSAR, Section [ ].~~
4. ~~FSAR, Section [ ].~~
5. ~~FSAR, Section [ ].~~
6. ~~6~~. CE NPSD-1045-A, "CEOG Joint Application Report for Modification to the Containment Spray System Technical Specifications," March 2000.
7. ~~7~~. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.
8. ~~8. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.~~
9. ~~9~~. ASME Code for Operation and Maintenance of Nuclear Power Plants.

St. Lucie Unit 1
Revision XXX

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6~~A~~ Containment Spray and Cooling Systems ~~(Atmospheric and Dual)~~ 1
~~(Credit taken for iodine removal by the Containment Spray System)~~

BASES

BACKGROUND

The Containment Spray and Containment Cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling systems are designed to the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1), or other documents that were appropriate at the time of licensing (identified on a unit specific basis).

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and the Containment Cooling System provide redundant methods to limit and maintain post accident conditions to less than the containment design values.

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray header~~s~~, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water tank (RWT) supplies borated water to the containment spray during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWT to the containment sump(s).

shutdown heat exchanger, 2

The Containment Spray System provides a spray of cold borated water ~~mixed with sodium hydroxide from the spray additive tank~~ into the upper regions of containment to reduce containment pressure and temperature and to reduce the concentration of fission products in the containment atmosphere during a DBA. The RWT solution temperature is an 8

BASES

BACKGROUND (continued)

important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the shutdown cooling heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet 50% of the system design requirements for containment heat removal ~~and 100% of the iodine removal design bases~~.

~~The Spray Additive System injects a hydrazine (N_2H_4) solution into the spray. The resulting alkaline pH of the spray enhances its ability to scavenge fission products from the containment atmosphere. The N_2H_4 added to the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.~~

The Containment Spray System is actuated either automatically by a containment High-High pressure signal coincident with a safety injection actuation signal (SIAS) or manually. An automatic actuation opens the containment spray ~~pump discharge~~ valves, starts the two Containment Spray System pumps, and begins the injection phase. The containment spray header isolation valves open upon a containment spray actuation signal. A manual actuation of the Containment Spray System is available on the main control board to begin the same sequence. The injection phase continues until an RWT level Low signal is received. The Low level for the RWT generates a recirculation actuation signal that aligns valves from the containment spray pump suction to the containment sump. The Containment Spray System in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

isolation

Containment Cooling System

redundant essential cooling headers through normally open valves

Two trains of containment cooling, each of sufficient capacity to supply 50% of the design cooling requirement, are provided. Two trains with two fan units each are supplied with cooling water from ~~a separate train of service water cooling~~. All four fans are required to furnish the design cooling capacity. ~~Air is drawn into the coolers through the fans and discharged to the steam generator compartments and pressurizer compartment.~~

component

BASES

BACKGROUND (continued)

Each train is supplied with component cooling water from redundant essential cooling headers through normally open valves.

SIAS

In post accident operation following a ~~containment cooling~~ actuation ~~signal (CCAS),~~ all four Containment Cooling System fans are designed to start automatically in slow speed. ~~Cooling is shifted from the chilled water cooled coils to the service water cooled coils.~~ The temperature of the ~~service~~ water is an important factor in the heat removal capability of the fan units.

component cooling water and intake cooling

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment temperature and pressure are the loss of coolant accident (LOCA) and the main steam line break (MSLB). The DBA LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System and one train of the Containment Cooling System being rendered inoperable.

43.43

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is ~~[55.7]~~ psig (experienced during a~~n~~ ~~MSLB~~). The analysis shows that the peak containment vapor temperature is ~~[413]~~ °F (experienced during an MSLB). Both results are within the design. (See the Bases for Specifications 3.6.4~~A and 3.6.4B~~, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of ~~[102]~~% RTP, one containment spray train and one containment cooling train operating, and initial (pre-accident) conditions of ~~[~~120~~]~~°F and ~~[14.7]~~ psia. The analyses also assume a response time delayed initiation in order to provide a conservative calculation of peak containment pressure and temperature responses.

LOCA

377

100.3

15.41

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation ~~reduces the~~ containment pressure ~~to [2.8] psig~~ due to the sudden cooling effect in the interior of the air tight containment. Additional discussion is provided in the Bases for Specifications 3.6.4~~A, 3.6.4B~~, and 3.6.~~12~~, "Vacuum Relief Valves."

results in an annulus to

of 0.9983 psid

differential

8

The modeled Containment Spray System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint coincident with an SIAS to achieve full flow through the containment spray nozzles. The

BASES

APPLICABLE SAFETY ANALYSES (continued)

Containment Spray System total response time of ~~[60]~~ seconds includes diesel generator startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 2).

The performance of the containment cooling train for post accident conditions is given in Reference ~~3~~. The result of the analysis is that each train can provide 50% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference ~~4~~.

The modeled Containment Cooling System actuation from the containment analysis is based upon the unit specific response time associated with exceeding the ~~CCAS~~ to achieve full Containment Cooling System air and safety grade cooling water flow.

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, a minimum of two containment cooling trains or two containment spray trains, or one of each, is required to maintain the containment peak pressure and temperature below the design limits (Ref. ~~5~~). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming that the worst case single active failure occurs.

Each Containment Spray System includes a spray pump, spray header~~s~~, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWT upon an ESF actuation signal and automatically transferring suction to the containment sump. Management of gas voids is important to Containment Spray System OPERABILITY.

Each Containment Cooling System includes ~~demisters,~~ cooling coils, ~~dampers,~~ fans, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, ~~3,~~ and ~~4~~, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the containment spray trains and containment cooling trains.

Containment spray trains are not required to be OPERABLE when pressurizer pressure is < 1750 psia.

BASES

APPLICABILITY (continued)

In MODES ~~5~~ and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the ~~Containment Spray and~~ Containment Cooling ~~systems are~~ not required to be OPERABLE in MODES ~~5~~ and 6.

ACTIONS

A.1

~~REVIEWER'S NOTE~~

~~Utilization of the 7 day Completion Time for Required Action A.1 is dependent on the licensee adopting CE NPSD-1045-A (Ref. 6) and meeting the requirements of the Topical Report and the associated Safety Evaluation including the following commitment: "[LICENSEE] has enhanced its Configuration Risk Management Program, as implemented under 10 CFR 50.65(a)(4), the Maintenance Rule, to include a Large Early Release Fraction assessment to support this application." Otherwise, a 72 hour Completion Time applies.~~

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within ~~[~~7~~]~~ days ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The ~~[~~7~~]~~ day Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and the findings of Ref. ~~6~~.

B.1

With one required containment cooling train inoperable, the inoperable containment cooling train must be restored to OPERABLE status within 7 days ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The components in this degraded condition are capable of providing greater than 100% of the heat removal needs (for the condition of one containment cooling train inoperable) after an accident. The 7 day Completion Time was developed based on the same reasons as those for Required Action A.1.

BASES

ACTIONS (continued)

C.1

and pressurizer pressure ≥ 1750 psia

With two required containment spray trains inoperable~~,~~ at least one of the required containment spray trains must be restored to OPERABLE status within 24 hours. Both trains of containment cooling must be OPERABLE or Condition ~~C~~ is also entered. The Condition is modified by a Note stating it is not applicable if the second containment spray train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. In addition, LCO 3.7.~~11~~, "~~CREACS~~," must be verified to be met within 1 hour. The components in this degraded condition are capable of providing greater than 100% of the heat removal needs after an accident. The Completion Time is based on Reference ~~7~~ which demonstrated that the 24 hour Completion Time is acceptable based on the redundant heat removal capabilities afforded by the Containment Cooling System, the iodine removal capability of ~~the Control Room Emergency Air Cleanup System~~, the infrequent use of the Required Action, and the small incremental effect on plant risk.

D.1 and D.2

with pressurizer pressure ≥ 1750 psia

With one containment spray train and one containment cooling train inoperable~~,~~ one of the required containment spray or containment cooling trains must be restored to OPERABLE status within 72 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System, the iodine removal function of the Containment Spray System, and the low probability of a DBA occurring during this period.

E.1

and pressurizer pressure ≥ 1750 psia

With two required containment cooling trains inoperable~~,~~ one of the required containment cooling trains must be restored to OPERABLE status within 72 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour

BASES

ACTIONS (continued)

Completion Time was developed taking into account the redundant heat removal capabilities afforded by ~~combinations of~~ the Containment Spray System ~~and Containment Cooling System~~, the iodine removal function of the Containment Spray System, and the low probability of a DBA occurring during this period.

F.1 and F.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

condition where the requirements do not apply. Containment spray trains are not required to be OPERABLE when pressurizer pressure is below 1750 psia and other Conditions apply when containment cooling trains are inoperable with pressurizer pressure below 1750 psia.

of Condition A, B, C, D, or E

If the Required Actions and associated Completion Times are not met, the plant must be brought to a ~~MODE in which overall plant risk is minimized~~. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and ~~to MODE 4~~ within 12 hours.

reduce pressurizer pressure to < 1750 psia

~~Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.~~

If one or more containment cooling trains remain inoperable once pressurizer pressure is < 1750 psia, the Conditions (i.e., Conditions G and I) must be entered, as applicable, and Required Actions performed within the associated Completion Times.

~~Required Action F.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the~~

BASES

ACTIONS (continued)

~~results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Insert 1

I

~~G~~.1

with pressurizer pressure ≥ 1750 psia

With any combination of three or more Containment Spray System and Containment Cooling System trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

Insert 2

SURVEILLANCE REQUIREMENTS

SR 3.6.6~~A~~.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This SR does not require any testing or valve manipulation. Rather, it involves verifying that those valves outside containment and capable of potentially being mispositioned are in the correct position.

~~[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

7

INSERT 1

G.1

With one required containment cooling train inoperable in MODE 3 with pressurizer pressure < 1750 psia, the inoperable containment cooling train must be restored to OPERABLE status within 72 hours. The remaining OPERABLE containment cooling train can provide sufficient heat removal needs after an accident when in MODE 3 and pressurizer pressure < 1750 psia. The 72 hours Completion Time was developed considering the reduced heat removal needs and the low probability of a DBA occurring during this period.

H.1

If the Required Action and associated Completion Time of Condition G are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 4 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant condition in an orderly manner.

7

INSERT 2

Additionally, with two containment cooling trains inoperable with pressurizer pressure < 1750 psia the containment cooling function may be lost since the Containment Spray System is not required to be OPERABLE below 1750 psia.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent flow path who is in continuous communication with the operators in the control room. This individual will have a method to rapidly close the system vent flow path if directed.

SR 3.6.6~~A~~.2

Operating each containment cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected and corrective action taken. ~~[The 31 day Frequency of this SR was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of a significant degradation of the containment cooling train occurring between surveillances and has been shown to be acceptable through operating experience.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6~~A~~.3

cooling

1200

Verifying a ~~service~~ water flow rate of ≥ ~~[2000]~~ gpm to each cooling unit provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 2). ~~[Also considered in selecting the Frequency of 31 days were the known reliability of the Cooling Water System, the two train redundancy, and the low probability of a significant degradation of flow occurring between surveillances.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~[~~ SR 3.6.6~~A~~.4

~~Verifying that the containment spray header piping is full of water to the [100] ft level minimizes the time required to fill the header. This ensures that spray flow will be admitted to the containment atmosphere within the time frame assumed in the containment analysis. [The 31 day Frequency is based on the static nature of the fill header and the low probability of a significant degradation of water level in the piping occurring between surveillances.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]]~~

This surveillance verifies that each automatic valve in the recirculation mode flow path actuates to the correct position on an actual or simulated actuation signal. The surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6~~A~~.5 1

Containment Spray System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the required containment spray trains and may also prevent water hammer and pump cavitation.

Selection of Containment Spray System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The Containment Spray System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the Containment Spray System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits. | R1

Containment Spray System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void

BASES

SURVEILLANCE REQUIREMENTS (continued)

volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the Containment Spray System piping and the procedural controls governing system operation.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.6.6~~A~~.6

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by the ASME Code (Ref. ~~9~~). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.6.6~~A~~.7 and SR 3.6.6~~A~~.8

These SRs verify that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~[The [18] month~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

The surveillance of containment sump isolation valves is also required by SR 3.5.2.~~5~~. A single surveillance may be used to satisfy both requirements.

SR 3.6.6~~A~~.9

This SR verifies that each containment cooling train actuates upon receipt of an actual or simulated actuation signal. ~~[The [18] month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6A.7 and SR 3.6.6A.8, above, for further discussion of the basis for the [18] month Frequency.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6~~A~~.10

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. Performance of this SR demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. ~~[~~Due to the passive design of the nozzle, a test ~~at [the first refueling and at] 10 year intervals~~ is considered adequate to detect obstruction of the spray nozzles following maintenance that could result in nozzle blockage.

Foreign material introduced as a result of maintenance, although unlikely, is the most likely potential cause for future spray nozzle obstruction. Therefore, verification conducted to confirm that potentially affected nozzles are free of blockage following maintenance activities that could result in nozzle blockage, is sufficient to confirm that the nozzles are free of obstruction.

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. FSAR, Section ~~[ ]~~.
3. ~~FSAR, Section [ ].~~
4. ~~FSAR, Section [ ].~~
5. ~~FSAR, Section [ ].~~
6. CE NPSD-1045-A, "CEOG Joint Application Report for Modification to the Containment Spray System Technical Specifications," March 2000.
7. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.
8. ~~CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.~~
9. ASME Code for Operation and Maintenance of Nuclear Power Plants.

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.6 BASES, CONTAINMENT SPRAY AND COOLING SYSTEMS

1. The type of Containment (Atmospheric and Dual), Specification designator “A” and the descriptor (credit taken for iodine removal by the Containment Spray System) are deleted since they are unnecessary (only one Containment Specification type is used in the St. Lucie (PSL) Plant ITS). ISTS 3.6.6A is used for PSL Unit 1 and Unit 2. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation.

2. Changes are made (additions, deletions, and/or changes) to the ITS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. The PSL design does not include Essential Chilled Water (ECW). Therefore, ISTS 3.7.10 is not included in the PSL ITS. The Control Room Emergency Air Cleanup System (ISTS 3.7.11) is renumbered as ITS 3.7.10.

6. The LCO 3.0.4.a NOTE is not applicable for the PSL Unit 1 and 2 and has been removed.

7. Concomitant Bases changes made to reflect changes to the ISTS Specification regarding change to Applicability and ACTIONS to include different Containment Spray and Containment Cooling System requirements in MODE 3 based on pressurizer pressure.

8. **Unit 2 only:** “St. Lucie Plant, Unit No. 2 – Issuance of Amendment No. 201 Regarding Technical Specification Changes to Eliminate the Requirements of the Iodine Removal System,” dated November 20, 2019 (ADAMS Accession No. ML19248C238) approved the elimination of Iodine Removal System requirements for the Technical Specifications. Hydrazine is not credited for maintaining post-LOCA sump pH, nor is it credited in managing post-accident iodine in the radiological consequences of design-basis accidents. Reference to the iodine removal system and hydrazine are removed.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.6.6, CONTAINMENT SPRAY AND COOLING SYSTEMS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

ITS 3.6.7, Shield Building

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

CONTAINMENT SYSTEMS

SHIELD BUILDING ~~INTEGRITY~~

LIMITING CONDITION FOR OPERATION

LCO 3.6.7 3.6.6.2 SHIELD BUILDING ~~INTEGRITY~~ shall be ~~maintained~~ OPERABLE.

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTION:

ACTION A Without SHIELD BUILDING INTEGRITY, restore SHIELD BUILDING INTEGRITY within 24 hours or be in at least HOT STANDBY within the next 6 hours and in HOT

ACTION B SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT

Required Action B.2 NOTE SHUTDOWN.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1 4.6.6.2 SHIELD BUILDING INTEGRITY shall be demonstrated in accordance with the Surveillance Frequency Control Program by verifying that the door in each access opening is closed except when the access opening is being used for normal transit

LCO 3.6.7 NOTE entry and exit.

CONTAINMENT SYSTEMS

SHIELD BUILDING ~~STRUCTURAL~~ INTEGRITY

LIMITING CONDITION FOR OPERATION

LCO 3.6.7

3.6.6.3 ~~The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.6.3.~~

A02

APPLICABILITY: ~~MODES 1, 2, 3 and 4.~~

ACTION:

~~With the structural integrity of the shield building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.~~

A03

SURVEILLANCE REQUIREMENTS

SR 3.6.7.2

4.6.6.3 The structural integrity of the shield building shall be determined, in accordance with the Containment Leakage Rate Testing Program, by a visual inspection of the accessible interior and exterior surfaces of the shield building ~~and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation~~.

LA01

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent shield building ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one shield building ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.6.9

NOTE
Action not applicable when second shield building ventilation system intentionally made inoperable.

b. With two shield building ventilation systems inoperable, within 1 hour verify at least one train of containment spray is OPERABLE, and restore at least one shield building ventilation system to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each shield building ventilation system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 continuous minutes with the heaters on.

See ITS 3.6.9

b. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

SR 3.6.7.3

c. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.

See ITS 5.5.8

2. Verifying that the filtration system starts automatically on a Containment Isolation Signal (CIS).

3. Verifying that the filter cooling makeup air and cross connection valves can be manually opened.

See ITS 3.6.9

SR 3.6.7.3

4. Verifying that ~~each~~ system produces a negative pressure of ≥ 2.0 inches W.G. in the annulus within 2 minutes after a Containment Isolation Signal (CIS).

L02 | R1

DEFINITIONS

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3020 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power to the CEA drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

See ITS Chapter 1.0

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

~~SHIELD BUILDING INTEGRITY~~

1.28 SHIELD BUILDING INTEGRITY shall exist when:

SR 3.6.7.1

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

See ITS

~~c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.~~

LA02

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 SITE BOUNDARY means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

See ITS Chapter 1.0

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

CONTAINMENT SYSTEMS

SHIELD BUILDING ~~INTEGRITY~~

LIMITING CONDITION FOR OPERATION

LCO 3.6.7 3.6.6.2 SHIELD BUILDING ~~INTEGRITY~~ shall be ~~maintained~~ OPERABLE.

Applicability **APPLICABILITY:** MODES 1, 2, 3, and 4.

ACTION:

ACTION A Without SHIELD BUILDING INTEGRITY, restore SHIELD BUILDING INTEGRITY within 24 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within

ACTION B the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Required Action B.2 NOTE

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1 4.6.6.2 SHIELD BUILDING INTEGRITY shall be demonstrated in accordance with the Surveillance Frequency Control Program by verifying that the door in each access opening is closed except when the access opening is being used for normal transit

LCO 3.6.7 NOTE entry and exit.

CONTAINMENT SYSTEMS

SHIELD BUILDING ~~STRUCTURAL~~ INTEGRITY

LIMITING CONDITION FOR OPERATION

LCO 3.6.7

3.6.6.3 ~~The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.6.3.~~

A02

APPLICABILITY: ~~MODES 1, 2, 3, and 4.~~

ACTION:

ACTION A

With the structural integrity of the shield building not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours ~~and in COLD SHUTDOWN within the following 30 hours.~~

L01

SURVEILLANCE REQUIREMENTS

SR 3.6.7.2

4.6.6.3 The structural integrity of the shield building shall be determined, in accordance with the Containment Leakage Rate Testing Program, by a visual inspection of the exposed accessible interior and exterior surfaces of the shield building ~~and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation~~.

LA01

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

2. Performing airflow distribution to HEPA filters and charcoal adsorbers in accordance with ASME N510-1989. The distribution shall be ± 20% of the average flow per unit.

See ITS 5.5.8

c. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

d. In accordance with the Surveillance Frequency Control Program by:

See ITS 3.6.9

1. Verifying that the system starts on a Unit 2 containment isolation signal and on a fuel pool high radiation signal.

2. Verifying that the filter cooling makeup and cross connection valves can be manually opened.

SR 3.6.7.3

3. Verifying that ~~each~~ system produces a negative pressure of greater than or equal to 2.0 inches WG in the annulus within 99 seconds after a start signal.

A04 | R1

4. Verifying that each system achieves a negative pressure of greater than 0.125 inch WG in the fuel storage building after actuation of a fuel storage building high radiation test signal.

See ITS 3.6.9

DEFINITIONS

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary-to-secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE – PURGING

1.24 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

See ITS Chapter 1.0

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3020 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power to the CEA drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

~~SHIELD BUILDING INTEGRITY~~

See ITS Chapter 1.0

1.28 SHIELD BUILDING INTEGRITY shall exist when:

See ITS 3.6.7

SR 3.6.7.1

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

See ITS 3.6.9

~~c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.~~

LA02

DISCUSSION OF CHANGES
ITS 3.6.7, SHIELD BUILDING

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.6.6.2 states “SHIELD BUILDING INTEGRITY shall be maintained.” CTS 3.6.6.3 requires the structural integrity of the shield building to be maintained as determined by a visual inspection in accordance with the Containment Leakage Rate Testing Program.

ITS 3.6.7 is the shield building specification. ITS LCO 3.6.7 requires the shield building to be OPERABLE. ITS 3.6.7 ACTION A requires when the shield building is inoperable to restore the shield building to OPERABLE status with 24 hours. ITS SR 3.6.7.2 requires performance of a visual inspection of the interior and exterior surfaces of the shield building. This changes the CTS by replacing the specific SHIELD BUILDING INTEGRITY definition and all references to it with the requirement for Shield Building OPERABILITY. Additionally, it changes the CTS by combining CTS 3.6.6.2 and CTS 3.6.6.3 into one specification.

The purpose of CTS 3.6.6.2 and CTS 3.6.6.3 is to provide requirements pertaining to shield building OPERABILITY. This portion of the change (combining CTS 3.6.6.2 and CTS 3.6.6.3) is acceptable because moving these requirements into one LCO, ITS 3.6.7 centralizes the requirements. The CTS 3.6.6.2 references to SHIELD BUILDING INTEGRITY have been deleted because the CTS definition of SHIELD BUILDING INTEGRITY in CTS 1.28 is incorporated into ITS 3.6.7 and ITS 3.6.9 and is no longer maintained as a separate definition in the ITS. ITS 3.6.7 requires that the shield building shall be OPERABLE. The definition of OPERABLE and the subsequent ITS 3.6.7 LCO, ACTIONS, and Surveillance Requirements are sufficient to encompass the applicable requirements of the CTS definition. This change removes any confusion that may exist between the definition and the specific requirements of the LCO and is a presentation preference consistent with NUREG-1432, Rev. 5.0. Because the aspects of SHIELD BUILDING INTEGRITY definition requirements are maintained within the Specifications of ITS, this change is considered acceptable.

This change is designated as administrative, because it does not result in technical changes to the CTS.

A03 Unit 1 CTS 3.6.6.3 does not provide an ACTION to take if the shield building is inoperable while in MODE 1, 2, 3, or 4; it only includes a requirement that the shield building be restored to OPERABLE status prior to increasing Reactor Coolant System temperature above 200°F (i.e., MODE 4). Although not explicitly stated, if these requirements are not met the shield building integrity

requirements are not met and the CTS 3.6.2.2 actions would apply. When the shield building is inoperable and not restored to an OPERABLE status within the specified Completion Time ITS 3.6.7 ACTION B requires the unit be in MODE 3 within 6 hours and MODE 4 within 12 hours. This changes the CTS by stating the ACTIONS within the Specification rather than deferring to CTS 3.6.2.2 and deletes the Action to restore the LCO prior to entering MODE 4. In addition, it combines CTS 3.6.6.2 and CTS 3.6.6.3 into one specification.

This change is acceptable because the ACTIONS specified in ITS 3.6.7 adopt ISTS structure for placing the unit outside the MODE of Applicability without changing the time specified to enter MODE 3 and MODE 4. In addition, deletion of the current Action of CTS 3.6.6.3 is acceptable because CTS 3.0.4 (ITS LCO 3.0.4) already precludes entering the MODE of Applicability when the LCO is not met. Therefore, it is not necessary to include these requirements as specific actions in ITS 3.6.7. This change is designated as administrative, because it does not result in technical changes to the CTS.

A04 **Unit 2 Only** CTS 4.6.1.1.d.3 requires a drawdown of the shield building annulus by each Shield Building Ventilation System (SBVS) train to within limits in accordance with the Surveillance Frequency Control Program. The Frequency in accordance with the Surveillance Frequency Control Program for PSL Unit 2 is “at least once per 36 months on a Staggered Test Basis.” ITS SR 3.6.7.3 requires a drawdown of the shield building to within limits using one SBVS train." R1

The purpose of CTS 4.6.6.1.c.4 is to verify the integrity of the shield building boundary by ensuring the shield building annulus can be rapidly drawn to a negative pressure of at least -2.0 inches water gauge. Staggering use of the SBVS train every 18 months (i.e., 36 months total) will ensure both trains are capable of producing the proper negative pressure in the shield building annulus. The specified Surveillance Frequency “on a Staggered Test Basis" continues to require each SBVS train to be tested. Therefore, this change is designated as administrative.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – Removal of LCO, SR, or other TS Requirements to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 4.6.6.3 requires the structural integrity of the shield building to be determined by a visual inspection of the accessible shield building interior and exterior surfaces and verifying no apparent changes in concrete surface appearance or other abnormal degradation. ITS SR 3.6.7.2 includes the shield building structural integrity visual inspection verification of exposed interior and exterior surfaces but does not

include the details of what the inspection entails. This changes the CTS by moving the details of the shield building inspection to the TS Bases.

The removal of these details, which are related to methods of surveillance test performance, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirements for verifying integrity of the shield building. Also, this changes is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specifications Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to methods of surveillance test performance is being removed from the Technical Specifications.

LA02 *(Type 2 – Removing Descriptions of System Operation)* CTS 1.28 states, in part, "SHIELD BUILDING INTEGRITY shall exist when: c. The sealing mechanism associated with each penetration (e.g., welds, bellows, or 0-rings) is OPERABLE. ITS 3.6.7 states "The shield building shall be OPERABLE." This changes the CTS by moving the reference to penetration sealing mechanism requirements to the Bases.

The removal of these details, which are related to system operation, from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for the shield building to be OPERABLE and the relocated material describes aspects of OPERABILITY. In addition, the ITS retains the requirement to perform a shield building annulus drawdown test, which would provide verification that the penetration sealing mechanisms are OPERABLE. Also, this change is acceptable, because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system operation is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 **Unit 2 Only** *(Category 4 – Relaxation of Required Action)* Unit 2 CTS 3.6.6.3, ACTION, states, "With the structural integrity of the shield building not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ITS 3.6.7 ACTION A requires that if the shield building is inoperable, it must be restored to OPERABLE status within 24 hours. ITS 3.6.7 is the shield building specification. ITS 3.6.7 combines CTS 3.6.6.2 and CTS 3.6.6.3 to address shield building OPERABILITY (see DOC A02). The end state of ITS 3.6.7 is MODE 4. ITS 3.6.7 ACTION B requires that if the Required Action and associated Completion Time

are not met (i.e., the shield building is not restored to OPERABLE status in 24 hours), the unit must be in MODE 3 within 6 hours and MODE 4 within 12 hours. In addition, ITS ACTION B includes a NOTE modifying Required Action B.2 that states, “LCO 3.0.4.a is not applicable when entering MODE 4.” This changes CTS by modifying the end state of CTS 3.6.6.3.

This change is acceptable because the modification of the end state from MODE 5 to MODE 4, with respect to shield building OPERABILITY, is consistent with CE NPSD-1186-A, Revision 00, “Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs,” dated October 2001 (ADAMS Accession No. 110410539) and Technical Specifications Task Force (TSTF) traveler TSTF- 422, Revision 2, “Change in Technical Specifications End States, (CE NPSD-1186),“ dated December 22, 2009 (ADAMS Accession No. ML103270197) allowances.

This change is designated as less restrictive because instead of requiring the plant to achieve COLD SHUTDOWN (MODE 5) the ITS end state is HOT SHUTDOWN (MODE 4) with LCO 3.0.4.a allowances applicable.

L02 Unit 1 Only (Category 7 – Relaxation of Surveillance Frequency) CTS 4.6.1.1.c.4 requires a drawdown of the shield building annulus by each Shield Building Ventilation System (SBVS) train to within limits in accordance with the Surveillance Frequency Control Program. The Frequency in accordance with the Surveillance Frequency Control Program for PSL Unit 1 is “at least once per 18 months.” ITS SR 3.6.7.3 requires a drawdown of the shield building to within limits using one SBVS train." The specified Surveillance Frequency in accordance with the Surveillance Frequency Control Program is proposed to be changed to "36 months on a Staggered Test Basis for each SBVS train” consistent with the ISTS 3.6.11.4 (ITS 3.6.7.3) Frequency as modified by the CTS 1.32 definition of “Staggered Test Basis." This change continues to require each SBVS train to be tested but changes the CTS by allowing the drawdown test for each SBVS train to be performed less frequently. R1

The purpose of CTS 4.6.6.1.c.4 is to verify the integrity of the shield building boundary by ensuring the shield building annulus can be rapidly drawn to a negative pressure of at least -2.0 inches water gauge. Therefore, this is a test of shield building integrity and does not need to be performed every 18 months using each SBVS train. Staggering use of the SBVS train every 18 months (i.e., 36 months total) will ensure both trains are capable of producing the proper negative pressure in the shield building annulus. This change is acceptable because performing the drawdown test using one train of SBVS every 18 months will adequately verify shield building integrity. OPERABILITY of SBVS will be maintained through the application of the requirements of ITS 3.6.9, Shield Building Ventilation System (SBVS). This change is designated as less restrictive because the shield building annulus drawdown Surveillance will be performed less frequently with each SBVS train under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.6 CONTAINMENT SYSTEMS

3.6.~~11~~ Shield Building ~~(Dual)~~

3.6.6.2 LCO 3.6.~~11~~ Shield building shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. Shield building inoperable.	A.1 Restore shield building to OPERABLE status.	24 hours
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
~~SR 3.6.11.1 Verify annulus negative pressure is > [5] inches water gauge.~~	~~[12 hours~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
1.28.a	SR 3.6.~~11~~.2 (7.1) Verify ~~one~~ (the) shield building access door in each access opening is closed. (, except when the access door is open for normal transit entry and exit)	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
4.6.6.3	SR 3.6.~~11~~.3 (7.2) Verify shield building structural integrity by performing a visual inspection of the exposed (accessible) interior and exterior surfaces of the shield building.	During shutdown for SR 3.6.1.1 Type A tests
4.6.6.1.c.4	SR 3. 6.~~11~~.4 (7.3) Verify the shield building can be maintained at a pressure equal to or more negative than ~~[~~ -0.25~~]~~ (-2.0) inch water gauge in the annulus by one Shield Building (Ventilation) ~~Exhaust Air Cleanup~~ System train with a ~~final~~ flow rate ~~≤ [ ]~~ (≥ 5400 and ≤ 6600) cfm within ~~[~~1~~]~~ (2) minute (s) after a start signal.	~~[[18] months on a STAGGERED TEST BASIS for each Shield Building Exhaust Air Cleanup System~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

CTS

3.6 CONTAINMENT SYSTEMS

3.6.~~11~~ Shield Building ~~(Dual)~~

3.6.6.2 LCO 3.6.~~11~~ Shield building shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. Shield building inoperable.	A.1 Restore shield building to OPERABLE status.	24 hours
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
~~SR 3.6.11.1 Verify annulus negative pressure is > [5] inches water gauge.~~	~~[12 hours~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.~~11~~.2	Verify ~~one~~ the shield building access door in each access opening is closed, except when the access door is open for normal transit entry and exit	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 3.6.~~11~~.3	Verify shield building structural integrity by performing a visual inspection of the accessible exposed interior and exterior surfaces of the shield building.	During shutdown for SR 3.6.1.1 Type A tests
SR 3. 6.~~11~~.4	Verify the shield building can be maintained at a pressure equal to or more negative than ~~[~~ -0.25~~]~~ -2.0 inch water gauge in the annulus by one Shield Building ~~Exhaust Air Cleanup~~ Ventilation System train with a ~~final~~ flow rate ≤ ~~[ ]~~ ≥ 5400 and ≤ 6600 cfm within ~~[1] minute~~ 99 seconds after a start signal.	~~[[18] months on a STAGGERED TEST BASIS for each Shield Building Exhaust Air Cleanup System~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.7, SHIELD BUILDING

1. The heading and title for ISTS 3.6.11 include the parenthetical expression (Dual). This identifying information is not included in the St. Lucie Plant (PSL) ITS. This information is provided in the NUREG to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. Therefore, necessary editorial changes were made. In addition, PSL design does not include the Hydrogen Mixing System (ISTS 3.6.9) and the Iodine Cleanup System (ISTS 3.6.10); the PSL Unit 2 design does not include the Spray Additive System (ISTS 3.6.7). Therefore, ISTS 3.6.9 and ISTS 3.6.10 are not included in the PSL ITS. The Spray Additive System (ISTS 3.6.7) is renumbered as ITS 3.6.10, the Shield Building Exhaust Air Cleanup System (ISTS 3.6.8) is renumbered as ITS 3.6.9 and the Shield Building (ISTS 3.6.11) is renumbered as ITS 3.6.7.

2. Changes are made (additions, deletions, and/or changes) to the ITS which reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. ISTS SR 3.6.11.1 requires a verification of annulus negative pressure is > [5] inches water gauge every 12 hours or in accordance with the Surveillance Frequency Control Program. PSL does not require a routine verification of static shield building annulus pressure. However, the ITS continues to require periodic verification that the shield building can be maintained at a pressure equal to or more negative than -2.0 inches water gauge in the annulus by one Shield Building Ventilation System (SBVS) train within the required time after a start signal consistent with current technical specifications. The Frequency specified for the shield building drawdown test continues to be in accordance with the Surveillance Frequency Control Program (i.e., at least every 18 months). Operational experience has shown that the frequency of the shield building drawdown test is adequate to detect degradation of shield building integrity such that the proper vacuum cannot be met and maintained within the required time. The current surveillance testing associated with the shield building is considered adequate to assure, pursuant to the requirements of 10 CFR 50.36(c)(3), that the necessary quality of the shield building is maintained, that facility operation will be within safety limits, and that the shield building limiting condition for operation will be met. As a result, ISTS SR 3.6.11.1 is not included in the ITS consistent with the PSL current licensing basis.

4. ISTS SR 3.6.11.2 requires verification that "one" access door in each shield building access opening is closed. However, PSL design consists of one door for each shield building access opening. Therefore, the Surveillance is changed to verify "the" shield building access door in each access opening closed, thereby reflecting the plant-specific design. There is no allowance in ISTS 3.6.11 for when a shield building access door is open for normal transit entry and exit. Therefore, an exception to the requirement that the access opening doors be closed is made to allow for normal transit entry and exit. Shield building entry may be required to perform Technical Specification (TS) Surveillances, as well as other activities on equipment inside the shield building that are required by TS or activities on equipment that support TS-required equipment. Normal transient entry and exit is not intended to preclude performing other activities (i.e., non TS-required activities) if the shield building is entered to perform an allowed activity. The basis of this exception is the assumption that the number of entries into the shield building is small while the unit is operating in MODES 1, 2, 3, and 4 and that the transit time during which a door is open will be

short. This allowance is acceptable due to the low probability of an event that could pressurize the shield building during the short time that the access door is expected to be open in MODES 1, 2, 3, and 4. This change from the ISTS is consistent with the current licensing basis as defined in CTS 1.28, definition of SHIELD BUILDING INTEGRITY, which provides this exception to the requirement for the door in each access opening to be closed.

5. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~11~~ Shield Building ~~(Dual)~~

BASES

BACKGROUND

The shield building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

Following a LOCA, the Shield Building ~~Exhaust Air Cleanup~~ System (SB~~EAC~~S) establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. A description of the SB~~EAC~~S is provided in the Bases for Specification 3.6.~~8~~, "Shield Building ~~Exhaust Air Cleanup~~ System (SB~~EAC~~S)." Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the SB~~EAC~~S.

APPLICABLE SAFETY ANALYSES

The design basis for shield building OPERABILITY is a large break LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive material from the primary containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analysis.

The shield building satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Shield building OPERABILITY must be maintained to ensure proper operation of the SB~~EAC~~S and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analysis.

APPLICABILITY

Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the primary containment following a LOCA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when a main steam line break, LOCA, or control element assembly ejection accident could release radioactive material to the primary containment atmosphere.

In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, shield building OPERABILITY is not required in MODE 5 or 6.

2 **INSERT 1**

The isolation devices for the penetrations in the shield building boundary are a part of the shield building leak tight barrier. To maintain the shield building boundary leak tight, the sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) are required to be OPERABLE.

BASES

ACTIONS

A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours.

Twenty-four hours is a reasonable Completion Time considering the limited leakage design of the containment and the low probability of a DBA occurring during this time period.

B.1 and B.2

~~---------- REVIEWER'S NOTE ----------~~
~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~
~~----------~~

4

If the shield building cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 1). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and

BASES

ACTIONS (continued)

establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

~~SR 3.6.11.1~~

~~Verifying that shield building annulus pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. [The 12 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.~~
~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.6.~~11~~.2

Each Maintaining shield building OPERABILITY requires verifying ~~one~~ the door in the access opening closed. ~~[An~~ access opening ~~may~~ contains one ~~inner and one outer~~ door~~, or in some cases, shield building access openings are shared such that a shield building barrier may have multiple inner or multiple outer doors. The intent is to not breach the shield building boundary at any time when the shield building boundary is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times.]~~ ~~However,~~ and all shield building access doors are normally kept closed, except when the access opening is being used for normal entry and exit for conveyance through the access opening. ~~or when maintenance is being performed on an access opening~~. ~~[The Frequency of 31 days is based on engineering judgment and is considered adequate in view of other indications of door status available to the operator.~~

~~OR~~

3 **INSERT 2**

Shield building entry may be required to perform Technical Specification (TS) Surveillances, as well as other activities on equipment inside the shield building that are required by TS or activities on equipment that support TS-required equipment. Normal transient entry and exit is not intended to preclude performing other activities (i.e., non-TS-required activities) if the shield building is entered to perform an allowed activity. This allowance is acceptable due to the low probability of an event that could pressurize the shield building during the short time that the access door is expected to be open.

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.6.~~11~~.3

~~This Surveillance would~~ give advance indication of gross deterioration of the concrete structural integrity of the shield building. The Frequency of this SR is the same as that of SR 3.6.1.1. The verification is done during shutdown and as part of Type A leakage tests associated with SR 3.6.1.1.

SR 3.6.~~11~~.4

The SB~~EAC~~S produces a negative pressure to prevent leakage from the building. SR 3.6.~~11~~.4 verifies that the shield building can be rapidly drawn down to ≥ [0.25] inch water. This test is used to ensure shield building boundary integrity. Establishment of this pressure is confirmed by SR 3.6.~~11~~.4, which demonstrates that the shield building can be drawn down to ≥ [0.25] inches of water ≤ 1 minute using one SB~~EAC~~S train. The time limit ensures that no significant quantity of radioactive material leaks from the shield building prior to developing the negative pressure. Since this SR is a shield building boundary integrity test, it does not need to be performed with each SB~~EAC~~S train. ~~[The SBEACS train used for this Surveillance is staggered to ensure that in addition to the requirements of LCO 3.6.11.4, either train will perform this test.]~~ The primary purpose of this SR is to ensure shield building integrity. The secondary purpose of this SR is to ensure that the SB~~EAC~~S being tested functions as designed. The inoperability of the SB~~EAC~~S train does not necessarily constitute a failure of this Surveillance relative to the shield building OPERABILITY. ~~[The 18 month Frequency is consistent with Regulatory Guide 1.52 (Ref. 2) guidance for functional testing of the ability of the SBEACS.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4

5

REFERENCES

1. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

~~2. Regulatory Guide 1.52, Revision [2].~~

2

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~11~~ Shield Building ~~(Dual)~~

BASES

BACKGROUND

The shield building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

Following a LOCA, the Shield Building ~~Exhaust Air Cleanup~~ System (SB~~EAC~~S) establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. A description of the SB~~EAC~~S is provided in the Bases for Specification 3.6.~~8~~, "Shield Building ~~Exhaust Air Cleanup~~ System (SB~~EAC~~S)." Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the SB~~EAC~~S.

APPLICABLE SAFETY ANALYSES

The design basis for shield building OPERABILITY is a large break LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive material from the primary containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analysis.

The shield building satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Shield building OPERABILITY must be maintained to ensure proper operation of the SB~~EAC~~S and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analysis.

APPLICABILITY

Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the primary containment following a LOCA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when a main steam line break, LOCA, or control element assembly ejection accident could release radioactive material to the primary containment atmosphere.

In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, shield building OPERABILITY is not required in MODE 5 or 6.

2 **INSERT 1**

The isolation devices for the penetrations in the shield building boundary are a part of the shield building leak tight barrier. To maintain the shield building boundary leak tight, the sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) are required to be OPERABLE.

BASES

ACTIONS

A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours.

Twenty-four hours is a reasonable Completion Time considering the limited leakage design of the containment and the low probability of a DBA occurring during this time period.

B.1 and B.2

~~---------- REVIEWER'S NOTE ----------~~
~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~
~~----------~~

4

If the shield building cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 1). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and

BASES

ACTIONS (continued)

establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

~~SR 3.6.11.1~~

~~Verifying that shield building annulus pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. [The 12 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.~~
~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

3

SR 3.6.~~11~~.2

7.1

Each Maintaining shield building OPERABILITY requires verifying ~~one~~ the door in the access opening closed. ~~[An~~ access opening ~~may~~ contains one ~~inner and one outer~~ door~~, or in some cases, shield building access openings are shared such that a shield building barrier may have multiple inner or multiple outer doors. The intent is to not breach the shield building boundary at any time when the shield building boundary is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times.]~~ ~~However,~~ and all shield building access doors are normally kept closed, except when the access opening is being used for normal entry and exit ~~or when maintenance is being performed on an access opening~~. ~~[The Frequency of 31 days is based on engineering judgment and is considered adequate in view of other indications of door status available to the operator.~~

Insert 2

for conveyance through the access opening.

1 3 3 4 6 4

~~OR~~

3 **INSERT 2**

Shield building entry may be required to perform Technical Specification (TS) Surveillances, as well as other activities on equipment inside the shield building that are required by TS or activities on equipment that support TS-required equipment. Normal transient entry and exit is not intended to preclude performing other activities (i.e., non-TS-required activities) if the shield building is entered to perform an allowed activity. This allowance is acceptable due to the low probability of an event that could pressurize the shield building during the short time that the access door is expected to be open.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.6.~~11~~.3

A visual inspection of the exposed accessible shield building interior and exterior surfaces and verification that no apparent changes in the appearance of concrete surfaces or other abnormal degradation will

~~This Surveillance would~~ give advance indication of gross deterioration of the concrete structural integrity of the shield building. The Frequency of this SR is the same as that of SR 3.6.1.1. The verification is done during shutdown and as part of Type A leakage tests associated with SR 3.6.1.1.

SR 3.6.~~11~~.4

The SB~~EAC~~S produces a negative pressure to prevent leakage from the building. SR 3.6.~~11~~.4 verifies that the shield building can be rapidly drawn down to ≥ [0.25] inch water. This test is used to ensure shield building boundary integrity. Establishment of this pressure is confirmed by SR 3.6.~~11~~.4, which demonstrates that the shield building can be drawn down to ≥ [0.25] inches of water ≤ 1 minute using one SB~~EAC~~S train. The time limit ensures that no significant quantity of radioactive material leaks from the shield building prior to developing the negative pressure. Since this SR is a shield building boundary integrity test, it does not need to be performed with each SB~~EAC~~S train. ~~[The SBEACS train used for this Surveillance is staggered to ensure that in addition to the requirements of LCO 3.6.11.4, either train will perform this test.]~~ The primary purpose of this SR is to ensure shield building integrity. The secondary purpose of this SR is to ensure that the SB~~EAC~~S being tested functions as designed. The inoperability of the SB~~EAC~~S train does not necessarily constitute a failure of this Surveillance relative to the shield building OPERABILITY. ~~[The 18 month Frequency is consistent with Regulatory Guide 1.52 (Ref. 2) guidance for functional testing of the ability of the SBEACS.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4

5

REFERENCES

1. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

~~2. Regulatory Guide 1.52, Revision [2].~~

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.7 BASES, SHIELD BUILDING

1. The heading and title for ISTS 3.6.11 include the parenthetical expression (Dual). This identifying information is not included in the St. Lucie Plant (PSL) ITS. This information is provided in the NUREG to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion, but serves no purpose in a plant-specific implementation. Therefore, necessary editorial changes were made. In addition, PSL design does not include the Hydrogen Mixing System (ISTS 3.6.9) and the Iodine Cleanup System (ISTS 3.6.10); the PSL Unit 2 design does not include the Spray Additive System (ISTS 3.6.7). Therefore, ISTS 3.6.9 and ISTS 3.6.10 are not included in the PSL ITS. The Spray Additive System (ISTS 3.6.7) is renumbered as ITS 3.6.10, the Shield Building Exhaust Air Cleanup System (ISTS 3.6.8) is renumbered as ITS 3.6.9 and the Shield Building (ISTS 3.6.11) is renumbered as ITS 3.6.7.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. Changes have been made to be consistent with changes made to the Specification.

4. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is changed to reflect the current licensing basis.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

6. There are no allowances in the LCO for a shield building access opening door to be open when maintenance is being performed on an access opening.

7. Changes are made to include details moved from the Current Technical Specifications to the Bases.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.6.7, SHIELD BUILDING

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 8

ITS 3.6.8, Vacuum Relief Valves

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

CONTAINMENT SYSTEMS

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

LCO 3.6.8

3.6.5.1 Two vacuum relief lines shall be OPERABLE.

Applicability

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

ACTION A
ACTION B

With one vacuum relief line inoperable, restore the vacuum relief line to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

4.6.5.1 Verify each vacuum relief line OPERABLE in accordance with the INSERVICE TESTING PROGRAM.

CONTAINMENT SYSTEMS

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

LCO 3.6.8 3.6.5 Two vacuum relief lines shall be OPERABLE.

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTION:

ACTION A
ACTION B With one vacuum relief line inoperable, restore the vacuum relief line to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1 4.6.5 Verify each vacuum relief line OPERABLE in accordance with the INSERVICE TESTING PROGRAM.

DISCUSSION OF CHANGES
ITS 3.6.8, VACUUM RELIEF VALVES

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.6 CONTAINMENT SYSTEMS

3.6.~~12~~ Vacuum Relief Valves ~~(Dual)~~

3.6.5.1 LCO 3.6.~~12~~ Two vacuum relief lines shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. One vacuum relief line inoperable.	A.1 Restore vacuum relief line to OPERABLE status.	72 hours
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.6.5.1 SR 3.6.~~12~~.1 Verify each vacuum relief line OPERABLE in accordance with the INSERVICE TESTING PROGRAM.	In accordance with the INSERVICE TESTING PROGRAM

CTS

3.6 CONTAINMENT SYSTEMS

3.6.~~12~~ Vacuum Relief Valves ~~(Dual)~~

3.6.5.1 LCO 3.6.~~12~~ Two vacuum relief lines shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. One vacuum relief line inoperable.	A.1 Restore vacuum relief line to OPERABLE status.	72 hours
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.6.5 SR 3.6.~~12~~.1	Verify each vacuum relief line OPERABLE in accordance with the INSERVICE TESTING PROGRAM.	In accordance with the INSERVICE TESTING PROGRAM

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.8, VACUUM RELIEF VALVES

1. The parenthetical expression (Dual) in ISTS 3.6.12 is deleted since it is unnecessary. This identifying information is not included in the St. Lucie Plant (PSL) Unit 1 and 2 ITS. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation. In addition, the PSL design does not include the Hydrogen Mixing System (ISTS 3.6.9) and the Iodine Cleanup System (ISTS 3.6.10); the PSL Unit 2 design does not include the Spray Additive System (ISTS 3.6.7). Therefore, ISTS 3.6.9 and ISTS 3.6.10 are not included in the PSL ITS. The Spray Additive System (ISTS 3.6.7) is renumbered as ITS 3.6.10, the Shield Building Exhaust Air Cleanup System (ISTS 3.6.8) is renumbered as ITS 3.6.9, the Shield Building (ISTS 3.6.11) is renumbered as ITS 3.6.7 and ISTS 3.6.12 is renumbered as ITS 3.6.8.

2. Changes are made (additions, deletions, and/or changes) to the ITS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~12~~ Vacuum Relief Valves ~~(Dual)~~
8
1

BASES

BACKGROUND

The vacuum relief valves protect the containment vessel against negative pressure (i.e., a lower pressure inside than outside). Excessive negative pressure inside containment can occur if there is an inadvertent actuation of ~~the Containment Cooling System or~~ the Containment Spray System. Multiple equipment failures or human errors are necessary to have inadvertent actuation. 2

The containment pressure vessel contains two 100% vacuum relief lines installed in parallel that protect the containment from excessive external loading. The vacuum relief lines are 24 inch penetrations that connect the shield building annulus to the containment. Each vacuum relief line is isolated by a pneumatically operated butterfly valve in series with a check valve located on the containment side of the penetration.

Each butterfly valve is actuated by a separate pressure controller that senses the differential pressure between the containment and the annulus. Each butterfly valve is provided with an air accumulator that allows the valve to open following a loss of instrument air.

The combined pressure drop at rated flow through either vacuum relief line will not exceed the containment pressure vessel design external pressure differential of ~~[0.65]~~ psig with any prevailing atmospheric pressure. 3

1.04

APPLICABLE SAFETY ANALYSES

Design of the vacuum relief lines involves calculating the effect of an inadvertent containment spray actuation that can reduce the atmospheric temperature (and hence pressure) inside containment (Ref. 1). Conservative assumptions are used for all the pertinent parameters in the calculation. For example, the minimum spray water temperature is assumed, as well as maximum initial containment temperature, maximum spray flow, all trains of spray operating, etc. The resulting containment pressure versus time is calculated, including the effect of the vacuum relief valves opening when their negative pressure setpoint is reached. It is also assumed that one vacuum relief line fails to open.

re-rated
Insert 1
2
1.04
The containment was ~~designed~~ for an external pressure load equivalent to ~~[0.65]~~ psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was ~~[-0.368] psig~~. This resulted in a ~~minimum pressure inside~~ containment of ~~[0.49] psig~~, which is less than the design load.
3
14.21 psia
0.9996 psid
maximum annulus to
differential pressure
2

2 **INSERT 1**

in accordance with paragraph NE-3133.3 of Section III of the ASME Boiler and Pressure Vessel Code (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The vacuum relief valves must also perform the containment isolation function in a containment high pressure event. For this reason, the system is designed to take the full containment positive design pressure and the containment design basis accident (DBA) environmental conditions (temperature, pressure, humidity, radiation, chemical attack, etc.) associated with the containment DBA.

The vacuum relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum equipment required to accomplish the vacuum relief function following the inadvertent actuation of the Containment Spray System. Two vacuum relief lines are required to be OPERABLE to ensure that at least one is available, assuming one or both valves in the other line fail to open.

APPLICABILITY

In MODES 1, 2, 3, and 4, the containment cooling features, such as the Containment Spray System, are required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside containment could occur whenever these systems are required to be OPERABLE due to inadvertent actuation of these systems. Therefore, the vacuum relief lines are required to be OPERABLE in MODES 1, 2, 3, and 4 to mitigate the effects of inadvertent actuation of the Containment Spray System or Containment Cooling System.

In MODES 5 and 6, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations of these MODES. The Containment Spray System and Containment Cooling System are not required to be OPERABLE in MODES 5 and 6. Therefore, maintaining OPERABLE vacuum relief lines is not required in MODE 5 or 6.

ACTIONS

A.1

With one of the required vacuum relief lines inoperable, the inoperable line must be restored to OPERABLE status within 72 hours. The specified time period is consistent with other LCOs for the loss of one train of a system required to mitigate the consequences of a LOCA or other DBA.

BASES

ACTIONS (continued)

B.1 and B.2

If the vacuum relief line cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.~~12~~.1

This SR references the INSERVICE TESTING PROGRAM, which establishes the requirement that inservice testing of the ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Boiler and Pressure Vessel Code and applicable Addenda (Ref. ~~2~~). Therefore, SR Frequency is governed by the INSERVICE TESTING PROGRAM.

REFERENCES

1. FSAR, Section ~~[~~6.2~~]~~.

~~2~~. ASME Code for Operation and Maintenance of Nuclear Power Plants.

2. ASME Boiler and Pressure Vessel Code, 1971 and applicable Addenda through Summer 1972, Section III, Division 1, Subsection NE.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~12~~ Vacuum Relief Valves ~~(Dual)~~ 1
8

BASES

BACKGROUND

The vacuum relief valves protect the containment vessel against negative pressure (i.e., a lower pressure inside than outside). Excessive negative pressure inside containment can occur if there is an inadvertent actuation of ~~the Containment Cooling System or~~ the Containment Spray System. 2
Multiple equipment failures or human errors are necessary to have inadvertent actuation.

The containment pressure vessel contains two 100% vacuum relief lines installed in parallel that protect the containment from excessive external loading. The vacuum relief lines are 24 inch penetrations that connect the shield building annulus to the containment. Each vacuum relief line is isolated by a pneumatically operated butterfly valve in series with a check valve located on the containment side of the penetration.

Each butterfly valve is actuated by a separate pressure controller that senses the differential pressure between the containment and the annulus. Each butterfly valve is provided with an air accumulator that allows the valve to open following a loss of instrument air.

The combined pressure drop at rated flow through either vacuum relief line will not exceed the containment pressure vessel design external pressure differential of ~~[0.65]~~ psig with any prevailing atmospheric pressure. 3
1.05

APPLICABLE SAFETY ANALYSES

Design of the vacuum relief lines involves calculating the effect of an inadvertent containment spray actuation that can reduce the atmospheric temperature (and hence pressure) inside containment (Ref. 1). Conservative assumptions are used for all the pertinent parameters in the calculation. For example, the minimum spray water temperature is assumed, as well as maximum initial containment temperature, maximum spray flow, all trains of spray operating, etc. The resulting containment pressure versus time is calculated, including the effect of the vacuum relief valves opening when their negative pressure setpoint is reached. It is also assumed that one vacuum relief line fails to open.

re-rated
Insert 1 2
The containment was ~~designed~~ for an external pressure load equivalent to ~~[0.65]~~ psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was ~~[-0.368] psig~~. This resulted in a ~~minimum pressure inside~~ containment of ~~[0.49] psig~~, which is less than the design load. 3
1.05
14.28 psia
1.0 psid
maximum annulus to
differential pressure 2

2 **INSERT 1**

in accordance with paragraph NE-3133.3 of Section III of the ASME Boiler and Pressure Vessel Code (Ref. 2). | R1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The vacuum relief valves must also perform the containment isolation function in a containment high pressure event. For this reason, the system is designed to take the full containment positive design pressure and the containment design basis accident (DBA) environmental conditions (temperature, pressure, humidity, radiation, chemical attack, etc.) associated with the containment DBA.

The vacuum relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum equipment required to accomplish the vacuum relief function following the inadvertent actuation of the Containment Spray System. Two vacuum relief lines are required to be OPERABLE to ensure that at least one is available, assuming one or both valves in the other line fail to open.

APPLICABILITY

In MODES 1, 2, 3, and 4, the containment cooling features, such as the Containment Spray System, are required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside containment could occur whenever these systems are required to be OPERABLE due to inadvertent actuation of these systems. Therefore, the vacuum relief lines are required to be OPERABLE in MODES 1, 2, 3, and 4 to mitigate the effects of inadvertent actuation of the Containment Spray System or Containment Cooling System.

In MODES 5 and 6, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations of these MODES. The Containment Spray System and Containment Cooling System are not required to be OPERABLE in MODES 5 and 6. Therefore, maintaining OPERABLE vacuum relief lines is not required in MODE 5 or 6.

ACTIONS

A.1

With one of the required vacuum relief lines inoperable, the inoperable line must be restored to OPERABLE status within 72 hours. The specified time period is consistent with other LCOs for the loss of one train of a system required to mitigate the consequences of a LOCA or other DBA.

BASES

ACTIONS (continued)

B.1 and B.2

If the vacuum relief line cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.~~12~~.1

This SR references the INSERVICE TESTING PROGRAM, which establishes the requirement that inservice testing of the ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Boiler and Pressure Vessel Code and applicable Addenda (Ref. ~~2~~). Therefore, SR Frequency is governed by the INSERVICE TESTING PROGRAM.

REFERENCES

1. FSAR, Section ~~[~~6.2~~]~~.

~~2~~. ASME Code for Operation and Maintenance of Nuclear Power Plants.

2. ASME Boiler and Pressure Vessel Code, 1971 and applicable Addenda through Summer 1972, Section III, Division 1, Subsection NE.

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.8 BASES, VACUUM RELIEF VALVES

1. The parenthetical expression (Dual) in ISTS 3.6.12 is deleted since it is unnecessary. This identifying information is not included in the St. Lucie Plant (PSL) Unit 1 and 2 ITS. This information is provided in NUREG-1432, Rev. 5.0, to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion but serves no purpose in a plant specific implementation. In addition, the PSL design does not include the Hydrogen Mixing System (ISTS 3.6.9) and the Iodine Cleanup System (ISTS 3.6.10). The PSL Unit 2 design does not include the Spray Additive System (ISTS 3.6.7). Therefore, ISTS 3.6.9 and ISTS 3.6.10 are not included in the PSL ITS. The Spray Additive System (ISTS 3.6.7) is renumbered as ITS 3.6.10, the Shield Building Exhaust Air Cleanup System (ISTS 3.6.8) is renumbered as ITS 3.6.9, the Shield Building (ISTS 3.6.11) is renumbered as ITS 3.6.7 and ISTS 3.6.12 is renumbered as ITS 3.6.8.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.6.8, VACUUM RELIEF VALVES

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 9

ITS 3.6.9, SHIELD BUILDING VENTILATION SYSTEM

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM (SBVS)

LIMITING CONDITION FOR OPERATION

LCO 3.6.9 3.6.6.1 Two ~~independent~~ ~~shield building ventilation systems~~ (SBVS trains) shall be OPERABLE. LA01

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTION:

ACTION A a. With one shield building ventilation ~~system~~ (train) inoperable, restore the inoperable ~~system~~ (train) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next

ACTION C 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION B NOTE

NOTE

Action not applicable when second shield building ventilation ~~system~~ (train) intentionally made inoperable.

ACTION B b. With two shield building ventilation ~~system~~s (train) inoperable, within 1 hour verify at least one train of containment spray is OPERABLE, and restore at least one shield

ACTION C building ventilation ~~system~~ to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. (train)

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each shield building ventilation ~~system~~ (train) shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by ~~initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and~~ verifying that the train operates for at least 15 continuous minutes with the heaters on. LA02

SR 3.6.9.1

SR 3.6.9.2 b. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

c. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989. See ITS 5.5.8

SR 3.6.9.3 2. Verify~~ing that the filtration system starts automatically on a Containment Isolation Signal (CIS)~~. Insert 1 L01 L03

~~3. Verifying that the filter cooling makeup air and cross connection valves can be manually opened.~~ L02

4. Verifying that each system produces a negative pressure of ≥ 2.0 inches W.G. in the annulus within 2 minutes after a Containment Isolation Signal (CIS). See ITS 3.6.7

INSERT 1 L01 L03

each SBVS train actuates on an actual or simulated containment isolation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.

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DEFINITIONS

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3020 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power to the CEA drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

See ITS Chapter 1.0

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

~~SHIELD BUILDING INTEGRITY~~

1.28 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

See ITS 3.6.7

LCO 3.6.9

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

See ITS 3.6.7

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 SITE BOUNDARY means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

See ITS Chapter 1.0

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM (SBVS)

LIMITING CONDITION FOR OPERATION

SBVS trains

LCO 3.6.9 3.6.6.1 Two ~~independent~~ ~~Shield Building Ventilation Systems~~ shall be OPERABLE.

LA01

Applicability **APPLICABILITY:** ~~At all times in~~ MODES 1, 2, 3, and 4.
~~In addition,~~ during movement of recently irradiated fuel assemblies ~~or during crane operations with loads over recently irradiated fuel assemblies in the Spent Fuel Storage Pool in MODES 5 and 6~~.

L04

A02

ACTION:

~~a. With the SBVS inoperable solely due to loss of the SBVS capability to provide design basis filtered air evacuation from the Spent Fuel Pool area, only ACTION c is required. If the SBVS is inoperable for any other reason, concurrently implement ACTION b and ACTION c.~~

train

ACTION A
ACTION C
b. (1) With one SBVS inoperable in MODE 1, 2, 3, or 4, restore the inoperable ~~system~~ to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

A02 | R1

ACTION B NOTE

NOTE
Action not applicable when second SBVS intentionally made inoperable.

ACTION B
(2) With both SBVSs inoperable, within 1 hour verify at least one train of containment spray is OPERABLE, and restore at least one SBVS to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

ACTION C

A02

ACTION A
c. (1) With one SBVS inoperable ~~in any MODE~~, restore the inoperable system to OPERABLE status within 7 days; otherwise, suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool ~~and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool~~.

ACTION D

L04

during movement of recently irradiated fuel assemblies

A02

ACTION D
(2) With both SBVS inoperable ~~in any MODE~~, immediately suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool ~~and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool~~.

L04

SURVEILLANCE REQUIREMENTS

train

4.6.6.1 Each Shield Building Ventilation ~~System~~ shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by ~~initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and~~ verifying that the system operates for at least 15 continuous minutes with the heaters on.

LA02

SR 3.6.9.1

See ITS 5.5.8 | R1

b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Performing a visual examination of SBVS in accordance with ASME N510-1989.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

2. Performing airflow distribution to HEPA filters and charcoal adsorbers in accordance with ASME N510-1989. The distribution shall be ± 20% of the average flow per unit.

See ITS 5.5.8

SR 3.6.9.2

c. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

d. In accordance with the Surveillance Frequency Control Program by:

SR 3.6.9.3

SR 3.6.9.4

1. Verify~~ing that the system starts on a Unit 2 containment isolation signal and on a fuel pool high radiation signal.~~

Insert 1

L01

L03

Insert 2

~~2. Verifying that the filter cooling makeup and cross connection valves can be manually opened.~~

L02

Insert 3

3. Verifying that each system produces a negative pressure of greater than or equal to 2.0 inches WG in the annulus within 99 seconds after a start signal.

See ITS 3.6.7

Insert 4

SR 3.6.9.5

4. Verifying that each system achieves a negative pressure of greater than 0.125 inch WG in the fuel storage building after actuation of a fuel storage building high radiation test signal.

INSERT 1

SR 3.6.9.3 Note

----------NOTE----------
Only required to be met in MODE 1, 2, 3, and 4.

A02

INSERT 2

L01

L03

SR 3.6.9.3 each SBVS train actuates on an actual or simulated containment isolation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.

INSERT 3

L01

L03

SR 3.6.9.4 Note

----------NOTE----------
Only required to be met during movement of recently irradiated fuel assemblies.

SR 3.6.9.4 Verify each SBVS train actuates in the fuel pool area ventilation mode on an actual or simulated fuel pool area high radiation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.

INSERT 4

A02

SR 3.6.9.5 Note

----------NOTE----------
Only required to be met during movement of recently irradiated fuel assemblies.

~~Page Deleted~~

DEFINITIONS

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary-to-secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE – PURGING

1.24 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

See ITS Chapter 1.0

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3020 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power to the CEA drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

~~SHIELD BUILDING INTEGRITY~~

1.28 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

See ITS 3.6.7

LCO 3.6.9

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

See ITS 3.6.7

DISCUSSION OF CHANGES
ITS 3.6.9, SHIELD BUILDING VENTILATION SYSTEM (SBVS)

ADMINISTRATIVE CHANGES

A01 In the conversion of the Plant (PSL) Unit 1 and 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 2 only** - CTS 3.6.6.1 Applicability requires the SBVS to be OPERABLE at all times in MODES, 1, 2, 3, and 4 and in addition, during movement of recently irradiated fuel assemblies or during crane operations with loads over recently irradiated fuel assemblies in the Spent Fuel Storage Pool in MODES 5 and 6. CTS 3.6.6.1 Action a clarifies that, “With the SBVS inoperable solely due to loss of the SBVS capability to provide design basis filtered air evacuation from the Spent Fuel Pool area, only ACTION-c is required. If the SBVS is inoperable for any other reason, concurrently implement ACTION-b and ACTION-c. ITS 3.6.9 Applicability requires the SBVS to be OPERABLE in MODES, 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies and does not include the information provided in CTS 3.6.6.1 Action a. This changes the CTS by eliminating certain wording (e.g., “at all times” and in MODES 5 and 6) because the information is unnecessary to comply with the Technical Specification requirements.

CTS 4.6.6.1.d.1 requires verifying that the SBVS starts on a Unit 2 containment isolation signal and on a fuel pool high radiation signal. CTS 4.6.6.1.d.4 requires, in effect, verifying each SBVS system (i.e., train) achieves the proper negative pressure in the fuel storage building following a high radiation actuation signal. ITS SRs 3.6.9.3, 3.6.9.4, and 3.6.9.5 incorporate these surveillance requirements and adds Notes to clarify when these SRs are required to be met based on their SBVS function (i.e., either in MODES 1, 2, 3, and 4 for the shield building ventilation mode or during activities involving recently irradiated fuel assemblies for the fuel pool area ventilation mode). This changes the CTS by adding clarification notes to ensure the appropriate SRs are applied.

The formatting of the ITS Applicability is in accordance with TSTF-GG-05-01, Writer’s Guide for Plant-Specific Improved Technical Specifications, dated June 2009 (ISTS Writer’s Guide) and it is understood that ITS 3.6.9 applies at all times in MODES 1, 2, 3, and 4. Additionally, due to the radioactive decay of irradiated fuel assemblies following plant operation, movement of recently irradiated fuel assemblies would only be possible within the first 72 hours following subcriticality during a reactor shutdown. Therefore, stating “in MODES 5 and 6” is unnecessary. Also, the ITS 3.6.9 format specifically clarifies the Conditions to ensure that when the SBVS is inoperable solely due to loss of the SBVS capability to provide design basis filtered air evacuation from the spent fuel pool area, the appropriate ACTIONS are applied. Therefore, the CTS action clarification is not required in the ITS.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.6.6.1 states that two "independent" shield building ventilation systems shall be OPERABLE. ITS 3.6.9 requires two shield building ventilation system trains to be OPERABLE but does not include the details of what constitutes OPERABILITY. This changes the CTS by moving the detail that the two systems must be "independent" to the Bases.

The removal of this detail, which is related to system design, from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement that two shield building ventilation system trains shall be OPERABLE. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change, because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.6.6.1.a requires each SBVS train to be operated for a specified time with the heaters on and specifies that flow through the HEPA filters and charcoal adsorbers be initiated from the control room. ITS SR 3.6.9.1 includes the surveillance to operate each SBVS train for a specified time with the heaters operating but does not include the requirement that flow through the HEPA filters and charcoal adsorbers be initiated from the control room. This changes the CTS by moving the procedural requirement that flow through the HEPA filters and charcoal adsorbers be initiated from the control room to the TS Bases.

The removal of these details, that are related to methods of surveillance test performance, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirements for operating each SBVS train for a specified time with the heaters operating. In addition, this change is acceptable because the removed

information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specifications Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to methods of surveillance test performance is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.6.6.1.c.2 and Unit 2 CTS 4.6.6.1.d.1 requires verification that the filtration system starts automatically on a Containment Isolation Signal (CIS). Unit 2 CTS 4.6.6.1.d.1 also requires verification that the system starts on a fuel pool high radiation signal. ITS SR 3.6.9.3 requires verification that each train actuates except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position. This changes the CTS by excluding those valves that are locked, sealed, or otherwise secured in the actuated position from verification. | R1

The purpose of Unit 1 CTS 4.6.6.1.c.2 and Unit 2 CTS 4.6.6.1.d.1 is to provide assurance that if an event occurred requiring actuation of the shield building ventilation system, then those valves and dampers requiring automatic actuation would actuate to their correct position. Those automatic valves and dampers that are locked, sealed, or otherwise secured in position are not required to actuate on a containment isolation signal in order to perform their safety function because they are already in the required position. Testing such valves and dampers would not provide any additional assurance of OPERABILITY. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L02 *(Category 5 – Deletion of Surveillance Requirement)* Unit 1 CTS 4.6.6.1.c.3 and Unit 2 CTS 4.6.6.1.d.2 require verification that the filter cooling makeup air and cross-connection valves can be manually opened. This changes the CTS by deleting verification of the ability to manually open the filter cooling makeup air and cross-connection valves.

The purpose of Unit 1 CTS 4.6.6.1.c.3 and Unit 2 CTS 4.6.6.1.d.2 is to provide assurance that cooling air flow can be provided through the filters for the individual train or cross connected flow manually. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to provide confidence that the equipment can perform its assumed safety function. Manual operation of these valve is not required to perform their safety function and testing of the valves does not provide any additional assurance of OPERABILITY. Valves that are required to actuate will continue to be tested. This change is designated as less restrictive because less stringent

Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L03 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.6.6.1.c.2 and Unit 2 CTS 4.6.6.1.d.1 require verification that the filtration system starts automatically on a "Containment Isolation Signal (CIS)." In addition, Unit 2 CTS 4.6.6.1.d.1 requires verification that each train also starts automatically on a "fuel pool high radiation signal." ITS SR 3.6.9.3 and ITS SR 3.6.9.4 require verification that the system actuates on "an actual or simulated containment isolation signal." This changes the Unit 1 and Unit 2 CTS by explicitly allowing the use of either an actual or simulated signal for the tests.

The purpose of Unit 1 CTS 4.6.6.1.c.2 and Unit 2 CTS 4.6.6.1.d.1 is to ensure the SBVS filtration train actuates upon receipt of a CIS and for Unit 2 includes receipt of a fuel pool high radiation signal. This change is acceptable because it has been determined that the current Surveillance Requirement acceptance criteria are not the only method that can be used for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual" or "simulated" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L04 **Unit 2 only:** (*Category 2 – Relaxation of Applicability*) CTS 3.6.6.1 Applicability requires, in part, that two SBVS be operable during crane operations with loads over recently irradiated fuel assemblies in the spent fuel storage pool in MODES 5 and 6. ITS 3.6.9 does not include an equivalent applicability. In addition, since the Applicability now does not require the LCO to be met during crane operations with loads over recently irradiated fuel assemblies in the spent fuel storage pool, the portion of CTS 3.6.6.1 Actions c.(1) and c.(2) to suspend movement of crane operations, has also been deleted.

The purpose of CTS 3.6.6.1 Applicability is to ensure that the SBVS is OPERABLE to provide filtration to the shield building annulus area to minimize the consequences of accidents and transients, including a fuel handling accident (FHA) involving recently irradiated fuel (i.e., an irradiated fuel assembly that has occupied part of a critical reactor core within the previous 72 hours) in the containment. Since the SBVS also provides filtration of the spent fuel pool area in the fuel handling building, the CTS 3.6.6.1 Applicability also ensures that the SBVS can provide filtration to the fuel pool area to minimize the consequences of an FHA involving recently irradiated fuel in the spent fuel storage pool. According to UFSAR Section 15.7.4.1.2, an FHA is initiated by the dropping of an irradiated fuel assembly either in the containment or in the fuel building. The FHA analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. It is improbable to move irradiated fuel within 72 hours of a plant shutdown (i.e., keff < 0.99) because of the physical time required to perform a controlled plant shutdown, cooldown and depressurize the RCS, and disassemble the reactor vessel to

access irradiated fuel to begin fuel movement and PSL procedurally restricts movement of irradiated fuel until 72 hours after shutdown. However, the ITS requires two trains of the SBVS to be OPERABLE to ensure that, in the unlikely event an FHA in the containment or spent fuel storage pool occurs within 72 hours of a unit shutdown, ventilation and filtration are provided to minimize the consequences of this improbable event.

Crane travel related requirements were relocated from the CTS in Amendment 134, ("St. Lucie Units 1 and 2 – Issuance of Amendments Regarding the Relocation of Spent Fuel Pool Crane Technical Specification Requirements (TAC Nos. MB5667 and MB5668)," dated April 28, 2004 (ADAMS Accession No. ML 040440111)) to the Updated Final Safety Analysis Report where the operational detail is controlled under 10 CFR 50.59 criteria. As cited in the NRC Safety Evaluation associated with Amendment 134, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," regulatory guidelines for control of heavy load lifts provide assurance of safe handling of heavy loads in areas where a load drop could impact stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. Section 5.1.1 of NUREG-0612 provides guidelines for reducing the likelihood of dropping heavy loads and provides criteria for establishing safe load paths; procedures for load-handling operations; training of crane operators; design, testing, inspection, and maintenance of cranes and lifting devices; and analyses of the impact of heavy load drops. The guidelines in Sections 5.1.2 through 5.1.6 address alternatives to either further reduce the probability of a load-handling accident or mitigate the consequences of heavy load drops. These alternatives include using a single-failure-proof crane to improve reliability through increased factors of safety and through redundancy or duality in certain active components. Criteria for design of single-failure-proof cranes are included in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants."

This change is acceptable because the possibility of damage to a recently irradiated fuel assembly as a consequence of mishandling components other than a recently irradiated fuel assembly is minimized by thorough training, detailed procedures and equipment design. The PSL crane design precludes the handling of heavy objects, such as shipping casks, over the spent fuel pool storage racks. Administrative controls prevent the movement of heavy loads over the cask pit whenever the cask pit rack is installed in the cask area of the spent fuel storage pool. In addition, the cask handling crane design meets the regulatory guidance for single-failure-proof cranes in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants" and NUREG-0612. Administrative controls that control the movement of light loads or prevent movement of light loads over recently irradiated fuel assemblies are similar to those used for control of heavy loads, to the extent practicable, as advised in NUREG-0612. Consequently, the possibility of dropping a load other than a recently irradiated fuel assembly and damaging of recently irradiated fuel assemblies in the containment or spent fuel storage pool is remote. Therefore, the Applicability of CTS 3.6.6.1 requiring the SBVS to be operable during crane operations with loads over recently irradiated fuel assemblies in the spent fuel storage pool in MODES 5 and 6 and related actions to suspend crane operations is not necessary to be included in the technical specifications and is removed.

DISCUSSION OF CHANGES
ITS 3.6.9, SHIELD BUILDING VENTILATION SYSTEM (SBVS)

This change is designated as less restrictive because the LCO requirements are applicable in fewer conditions than in the CTS and the associated Required Actions are less stringent in the ITS than applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.6 CONTAINMENT SYSTEMS

3.6.~~8~~ Shield Building ~~Exhaust Air Cleanup~~ System (SB~~EAC~~S) ~~(Dual)~~

3.6.6.1 LCO 3.6.~~8~~ Two SB~~EAC~~S trains shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION a. A. One SB~~EAC~~S train inoperable.	A.1 Restore train to OPERABLE status.	7 days
ACTION b. B. ---------- NOTE ---------- Not applicable when second SB~~EAC~~S train intentionally made inoperable. ---------- Two SB~~EAC~~S trains inoperable.	B.1 Verify at least one train of containment spray is OPERABLE. AND B.2 Restore at least one SB~~EAC~~S train to OPERABLE status.	1 hour 24 hours
ACTION a. and b. C. Required Action and Associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.6.6.1.a	SR 3.6.~~8~~.1	Operate each SB~~EAC~~S train for ≥ 15 continuous minutes ~~[~~with heaters operating~~]~~.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
4.6.6.1.b	SR 3.6.~~8~~.2	Perform required SB~~EAC~~S filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
4.6.6.1.c.2	SR 3.6.~~8~~.3	Verify each SB~~EAC~~S train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
	~~SR 3.6.8.4~~	~~[Verify each SBEACS filter bypass damper can be opened, except for dampers that are locked, sealed, or otherwise secured in the open position.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.8.5 Verify each SBEACS train flow rate is ≥ [ ] cfm.	[[18] months on a STAGGERED TEST BASIS OR In accordance with the Surveillance Frequency Control Program]

4

CTS

3.6 CONTAINMENT SYSTEMS

3.6.8 Shield Building Exhaust Air Cleanup System (SBEACS) (Dual)

3.6.6.1 LCO 3.6.8 Two SBEACS trains shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION b.(1) and c.(1) A. One SBEACS train inoperable.	A.1 Restore train to OPERABLE status.	7 days
ACTION NOTE B. ---------- NOTE ---------- Not applicable when second SBEACS train intentionally made inoperable. ----------	B.1 Verify at least one train of containment spray is OPERABLE. AND	1 hour
ACTION b.(2) Two SBEACS trains inoperable.	B.2 Restore at least one SBEACS train to OPERABLE status.	24 hours
ACTION b.(1) and b.(2) C. Required Action and Associated Completion Time not met of Condition A or B in MODE 1, 2, 3, or 4 DOC A02	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

Insert 1

Insert 2

Insert 3

INSERT 1 2

Applicability During movement of recently irradiated fuel assemblies.

INSERT 2 2

DOC A02 in MODE 1, 2, 3, or 4

INSERT 3 2

ACTION c.(1) and c.(2) DOC A02	D. Required Action and associated Completion Time of Condition A not met during movement of recently irradiated fuel assemblies. OR Two SBVS trains inoperable during movement of recently irradiated fuel assemblies.	D.1 Suspend movement of recently irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.6.6.1.a	SR 3.6.~~8~~.1 Operate each SB~~EAC~~S train for ≥ 15 continuous minutes ~~[~~with heaters operating~~]~~.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
4.6.6.1.c	SR 3.6.~~8~~.2 Perform required SB~~EAC~~S filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
4.6.6.1.d.1	SR 3.6.~~8~~.3 Verify each SB~~EAC~~S train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position. containment isolation DOC A02 ----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4. ----------	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
DOC A02 4.6.6.1.d.1	SR 3. 6.~~8~~.4 ~~[Verify each SBEACS filter bypass damper can be opened, except for dampers that are locked, sealed, or otherwise secured in the open position.~~ ---------- NOTE ---------- Only required to be met during movement of recently irradiated fuel assemblies. ---------- Verify each SBVS train actuates in the fuel pool area ventilation mode on an actual or simulated fuel pool area high radiation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.8.5 Verify each SBEACS train flow rate is ≥ [ ] cfm. ----------NOTE---------- Only required to be met during movement of recently irradiated fuel assemblies. ---------- Verify each SBVS train can maintain a negative pressure ≥ 0.125 inches water gauge in the fuel handling building after actuation of a fuel pool area high radiation signal.	[[18] months on a STAGGERED TEST BASIS OR In accordance with the Surveillance Frequency Control Program]

DOC A02

4.6.6.1.d.4

R1

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.9, SHIELD BUILDING VENTILATION SYSTEM (SBVS)

1. The ISTS 3.6.8 title "Shield Building Exhaust Air Cleanup System (SBEACS)" has been changed to "Shield Building Ventilation System (SBVS)" consistent with the St. Lucie Plant (PSL) site specific terminology. The heading for ISTS 3.6.8 includes the parenthetical expression (Dual). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Hydrogen Mixing System (ISTS 3.6.9) and the Iodine Cleanup System (ISTS 3.6.10)); the PSL Unit 2 design does not include the Spray Additive System (ISTS 3.6.7). Therefore, ISTS 3.6.9 and ISTS 3.6.10 are not included in the PSL ITS. The Spray Additive System (ISTS 3.6.7) is renumbered as ITS 3.6.10, the Shield Building Exhaust Air Cleanup System (ISTS 3.6.8) is renumbered as ITS 3.6.9, the Shield Building (ISTS 3.6.11) is renumbered as ITS 3.6.7 and Vacuum Relief Valves (ISTS 3.6.12) is renumbered as ITS 3.6.8.

2. Changes are made (additions, deletions, and/or changes) to the ITS which reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. ISTS SR 3.6.8.4 and ISTS 3.6.8.5 contain bracketed information and/or values that are generic to all Combustion Engineering vintage plants. ISTS SR 3.6.8.4 requires verification in accordance with the Surveillance Frequency Control Program that the filter bypass damper can be opened. ISTS 3.6.8.5 requires verification of SBVS train flow rate in accordance with the Surveillance Frequency Control Program. PSL does not have filter bypass dampers. Therefore, this SR is deleted. The flow rate of each SBVS train required by ISTS SR 3.6.8.5 is verified as required by the Ventilation Filter Test Program (VFTP). ISTS 3.6.8.2 (ITS SR 3.6.9.2) requires performance of filter testing in accordance with the VFTP. Therefore, this SR is duplicative and not adopted into the PSL ITS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~8~~ Shield Building ~~Exhaust Air Cleanup~~ System (~~SBEACS~~) ~~(Dual)~~

BASES

BACKGROUND

The ~~SBEACS~~ is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive material leaking from the primary containment of a dual containment into the shield building (secondary containment) following a Design Basis Accident are filtered and adsorbed prior to exhausting to the environment.

The containment has a secondary containment, the shield building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

Following a LOCA, the ~~SBEACS~~ establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the ~~SBEACS~~.

The ~~SBEACS~~ consists of two separate and redundant trains. Each train includes ~~a heater~~, ~~cooling coils,~~ a prefilter, a moisture separator, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of radioiodines, and a fan. Ductwork, valves and/or dampers, and instrumentation also form part of the system. The moisture separator~~s~~ function to reduce the moisture content of the airstream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a ~~safety injection actuation~~ signal (~~SIAS~~). The system is described in Reference 2.

The prefilters remove any large particles in the air, and the moisture separator~~s~~ remove any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Heaters ~~may be~~ included to reduce the relative humidity of the airstream on

BASES

BACKGROUND (continued)

systems operating in high humidity. ~~[The cooling coils cool the air to keep the charcoal beds from becoming too hot due to absorption of fission products.]~~ 3

~~During normal operation, the Shield Building Cooling System is aligned to bypass the SBEACS HEPA filters and charcoal adsorbers. For SBEACS operation following a DBA, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers.~~ 2

SBVS

The ~~SBEACS~~ reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the ~~SBEACS~~ could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis. 1

APPLICABLE SAFETY ANALYSES

SBVS

The ~~SBEACS~~ design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) assumes that only one train of the ~~SBEACS~~ is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA. 1

SBVS

The modeled ~~SBEACS~~ actuation in the safety analysis is based on a worst case response time associated with exceeding a~~n SIAS~~. The total response time from exceeding the signal setpoint to attaining the negative pressure of ~~[≥ 0.25]~~ inch water gauge in the shield building is ~~[1 minute]~~. This response time is composed of signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting. 1 2 3

CIS

2.0

es

2 minutes

SBVS

The ~~SBEACS~~ satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

SBVS

In the event of a DBA, one ~~SBEACS~~ train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two trains of the ~~SBEACS~~ must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure. 1 | R1

SBVS

independent 6

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the Filtration System is not required to be OPERABLE.

ACTIONS

A.1

With one ~~SBEACS~~ train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant ~~SBEACS~~ train and the low probability of a DBA occurring during this period.

B.1 and B.2

If two ~~SBEACS~~ trains are inoperable, at least one ~~SBEACS~~ train must be returned to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second ~~SBEACS~~ train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. In addition, at least one train of containment spray must be verified to be OPERABLE within 1 hour. In the event of an accident, containment spray reduces the potential radioactive release from the containment, which reduces the consequences of the inoperable ~~SBEACS~~ trains. The Completion Time is based on Reference 4 which demonstrated that the 24 hour Completion Time is acceptable based on the infrequent use of the Required Actions and the small incremental effect on plant risk.

ACTIONS (continued)

C.1 and C.2

SBVS

If the ~~SBEACS~~ train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

1

SURVEILLANCE REQUIREMENTS

9

SR 3.6.~~8~~.1

SBVS

Operating each ~~SBEACS~~ train for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. ~~Experience from filter testing at operating units indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters.~~ ~~[The 31 day Frequency was developed considering the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System.~~

This SR is initiated from the control room with flow through the HEPA filter and charcoal adsorber train.

1

6 | R1

4

3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

5

3

SURVEILLANCE REQUIREMENTS (continued)

9

SR 3.6.~~8~~.2

1

SBVS

This SR verifies that the required ~~SBEACS~~ filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing of HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

9

SR 3.6.~~8~~.3

1

SBVS

The automatic startup ensures that each ~~SBEACS~~ train responds properly. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the SBEACS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.8.1.~~

3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

] 5 3

[SR 3.6.8.4

The filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal operation and must reposition for accident operation to draw air through the filters. The SR excludes automatic dampers that are locked, sealed, or otherwise secured in the open position. The SR does not apply to dampers that are locked, sealed, or otherwise secured in the open position since the affected dampers were verified to be in the open position prior to being locked, sealed, or otherwise secured. Placing an automatic damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the damper to be closed to support the accident analysis. Restoration of an automatic damper to the closed position requires verification that the SR has been met within its required Frequency. [The [18] month Frequency is considered to be acceptable based on the damper reliability and design, the mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the Surveillance when performed at the [18] month Frequency.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

]] 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.6.8.5~~

~~The SBEACS train flow rate is verified ≥ [ ] cfm to ensure that the flow rate is adequate to "pull down" the shield building pressure as required. This test also will verify the proper functioning of the fans, dampers, filters, absorbers, etc., when this SR is performed in conjunction with SR 3.6.11.4.~~

~~[The [18] month on a STAGGERED TEST BASIS Frequency is consistent with the Regulatory Guide 1.52 (Ref. 5) guidance.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 3

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. FSAR, Section ~~[ ]~~. 6.2 U
3. FSAR, Section ~~[ ]~~. 15
4. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

~~5. Regulatory Guide 1.52, Revision [2].~~ 3 2

3 2

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~8~~ Shield Building ~~Exhaust Air Cleanup~~ System (~~SBEACS~~) ~~(Dual)~~

BASES

BACKGROUND

The ~~SBEACS~~ is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive material leaking from the primary containment of a dual containment into the shield building (secondary containment) following a Design Basis Accident are filtered and adsorbed prior to exhausting to the environment.

The containment has a secondary containment, the shield building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

Following a LOCA, the ~~SBEACS~~ establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the ~~SBEACS~~.

The ~~SBEACS~~ consists of two separate and redundant trains. Each train includes ~~a heater~~, ~~cooling coils,~~ a prefilter, a moisture separator, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of radioiodines, and a fan. Ductwork, valves and/or dampers, and instrumentation also form part of the system. The moisture separator~~s~~ function to reduce the moisture content of the airstream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a ~~safety injection actuation~~ signal (~~SIAS~~). The system is described in Reference 2.

The prefilters remove any large particles in the air, and the moisture separator~~s~~ remove any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Heaters ~~may be~~ included to reduce the relative humidity of the airstream on

INSERT 1 2

A secondary function of the SBVS is to align and filter airborne radioactive particulate from the area of the fuel storage pool following a fuel handling accident.

INSERT 2 2

When aligned to evacuate the fuel handling building spent fuel pool area, the SBVS trains include two motor operated butterfly valves positioned to exhaust the fuel pool area.

INSERT 3 2

On a high fuel pool area radiation signal, the SBVS starts and aligns to exhaust the spent fuel pool area through the SBVS filtration units. In this alignment, the system achieves and maintains a negative air pressure in the fuel handling building of greater than 0.125 inch water gauge.

BASES

BACKGROUND (continued)

systems operating in high humidity. ~~[The cooling coils cool the air to keep the charcoal beds from becoming too hot due to absorption of fission products.]~~ 3

Insert 4

~~During normal operation, the Shield Building Cooling System is aligned to bypass the SBEACS HEPA filters and charcoal adsorbers. For SBEACS operation following a DBA, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers.~~ 2

SBVS

The ~~SBEACS~~ reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the ~~SBEACS~~ could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis. 1

Insert 5 2

APPLICABLE SAFETY ANALYSES

SBVS

The ~~SBEACS~~ design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) assumes that only one train of the ~~SBEACS~~ is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA. 1

SBVS

The modeled ~~SBEACS~~ actuation in the safety analysis is based on a worst case response time associated with exceeding a~~n SIAS~~. The total response time from exceeding the signal setpoint to attaining the negative pressure of ~~[≥ 0.25]~~ inch water gauge in the shield building is ~~[1 minute]~~. This response time is composed of signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting. 1 2 3

CIAS

2.0

es

99 seconds

SBVS

Insert 6 2

The ~~SBEACS~~ satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

SBVS

In the event of a DBA, one ~~SBEACS~~ train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two trains of the ~~SBEACS~~ must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure. 1 R1

SBVS

independent 6 2

Insert 7

INSERT 4 2

The SBVS is interconnected to the spent fuel pool area exhaust duct. Upon receipt of a high radiation signal in the spent fuel pool area, the exhaust is directed to the SBVS filtration units. Two fuel pool area interconnecting valves open and the SBVS fans start automatically. Two shield building valves isolate the annulus. In the event of a LOCA, a CIAS overrides the fuel pool area high radiation signal and initiates the depressurization of the shield building annulus.

INSERT 5 2

The SBVS when aligned to evacuate the fuel pool area is designed to limit offsite exposures resulting from a fuel handling accident involving recently irradiated fuel.

INSERT 6 2

The SBVS alignment to evacuate the fuel pool area is designed to mitigate the consequences of a fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 72 hours) in which rods in the fuel assembly are assumed to be damaged. The analysis of a fuel handling accident is given in Reference 3.

INSERT 7 2

The SBVS is considered OPERABLE for fuel handling building evacuation when the individual components necessary to evacuate the fuel pool area are OPERABLE in both SBVS trains.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

Insert 8
2

In MODES 5 and 6, ~~the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the Filtration System is not required to be OPERABLE.~~

ACTIONS

A.1

SBVS

With one ~~SBEACS~~ train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant ~~SBEACS~~ train and the low probability of a DBA occurring during this period.

1

1

SBVS

B.1 and B.2

Insert 9
2

SBVS

If two ~~SBEACS~~ trains are inoperable, at least one ~~SBEACS~~ train must be returned to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second ~~SBEACS~~ train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. In addition, at least one train of containment spray must be verified to be OPERABLE within 1 hour. In the event of an accident, containment spray reduces the potential radioactive release from the containment, which reduces the consequences of the inoperable ~~SBEACS~~ trains. The Completion Time is based on Reference 4 which demonstrated that the 24 hour Completion Time is acceptable based on the infrequent use of the Required Actions and the small incremental effect on plant risk.

1

SBVS

SBVS

1

INSERT 8 2

the shield building is not required to be OPERABLE. Therefore, the shield building ventilation mode of SBVS is not required to be OPERABLE.

During movement of recently irradiated fuel assemblies, the fuel pool area ventilation mode of SBVS is required to be OPERABLE to mitigate the consequences of a fuel handling accident involving handling recently irradiated fuel. Due to radioactive decay, SBVS is only required to mitigate fuel handling accidents involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 72 hours).

INSERT 9 2

in MODE 1, 2, 3, or 4

ACTIONS (continued)

C.1 and C.2

If the ~~SBEACS~~ SBVS train cannot be restored to OPERABLE status within the required Completion Time of Condition A or B when in Mode 1, 2, 3, or 4, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Insert 10

SURVEILLANCE REQUIREMENTS

SR 3.6.~~8~~9.1

Operating each ~~SBEACS~~ SBVS train for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. This SR is initiated from the control room with flow through the HEPA filter and charcoal adsorber train. ~~Experience from filter testing at operating units indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters. [The 31 day Frequency was developed considering the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

INSERT 10 2

D.1

When Required Action A.1 cannot be completed within the required Completion Time or both SBVS trains are inoperable due to an inability to evacuate the fuel pool area during movement of recently irradiated fuel assemblies, this operation must be suspended. Action must be taken to place the unit in a condition in which the fuel pool area ventilation mode of SBVS does not apply. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS (continued)

9

SR 3.6.~~8~~.2

1

SBVS

This SR verifies that the required ~~SBEACS~~ filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing of HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

9

SR 3.6.~~8~~.3

2

on an automatic CIAS

1

SBVS

The automatic startup ensures that each ~~SBEACS~~ train responds properly. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the SBEACS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.8.1.~~

3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Insert 11

2

INSERT 11 2

This SR is modified by a Note that states that the SR is only required to be met in MODES 1, 2, 3, and 4 because the shield building ventilation mode of SBVS is only required in these MODES.

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

5
3

~~[~~ SR 3.6.~~8~~.4

9
1
2
Insert 12

~~The filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal operation and must reposition for accident operation to draw air through the filters. The SR excludes automatic dampers that are locked, sealed, or otherwise secured in the open position. The SR does not apply to dampers that are locked, sealed, or otherwise secured in the open position since the affected dampers were verified to be in the open position prior to being locked, sealed, or otherwise secured. Placing an automatic damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the damper to be closed to support the accident analysis. Restoration of an automatic damper to the closed position requires verification that the SR has been met within its required Frequency. [The [18] month Frequency is considered to be acceptable based on the damper reliability and design, the mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the Surveillance when performed at the [18] month Frequency.~~

3

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]]~~

2

INSERT 12

This SR verifies that each SBVS train starts and operates on an actual or simulated high radiation actuation signal in the fuel pool area exhaust. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

A Note to the SR indicates that this test is only required to be met during movement of recently irradiated fuel assemblies because the fuel pool area ventilation mode of SBVS is only required during this operation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.~~8~~.5 9

Insert 13

~~The SBEACS train flow rate is verified ≥ [ ] cfm to ensure that the flow rate is adequate to "pull down" the shield building pressure as required. This test also will verify the proper functioning of the fans, dampers, filters, absorbers, etc., when this SR is performed in conjunction with SR 3.6.11.4.~~

~~[The [18] month on a STAGGERED TEST BASIS Frequency is consistent with the Regulatory Guide 1.52 (Ref. 5) guidance.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

1, 2, 3

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. FSAR, Section ~~[ ]~~. 6.2
3. FSAR, Section ~~[ ]~~. U, 15
4. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

~~5. Regulatory Guide 1.52, Revision [2].~~

3, 2

2

INSERT 13

This SR verifies proper operation of the SBVS and the ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas. The system is designed to maintain a slight negative pressure in the fuel pool area after actuation of a fuel pool area high radiation signal to prevent unfiltered release of radioactivity.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

A Note to the SR indicates that this test is only required to be met during movement of recently irradiated fuel assemblies because the fuel pool area ventilation mode of SBVS is only required during this operation.

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.9 BASES, SHIELD BUILDING VENTILATION SYSTEM (SBVS)

1. The ISTS 3.6.8 title "Shield Building Exhaust Air Cleanup System (SBEACS)" has been changed to "Shield Building Ventilation System (SBVS)" consistent with the St. Lucie Plant (PSL) site specific terminology. The heading for ISTS 3.6.8 includes the parenthetical expression (Dual). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, the PSL design does not include the Hydrogen Mixing System (ISTS 3.6.9) and the Iodine Cleanup System (ISTS 3.6.10); the PSL Unit 2 design does not include the Spray Additive System (ISTS 3.6.7). Therefore, ISTS 3.6.9 and ISTS 3.6.10 are not included in the PSL ITS. The Spray Additive System (ISTS 3.6.7) is renumbered as ITS 3.6.10, the Shield Building Exhaust Air Cleanup System (ISTS 3.6.8) is renumbered as ITS 3.6.9, the Shield Building (ISTS 3.6.11) is renumbered as ITS 3.6.7 and Vacuum Relief Valves (ISTS 3.6.12) is renumbered as ITS 3.6.8.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is changed to reflect the current licensing basis.

4. Changes are made to the ITS Bases consistent and in conformance with St. Lucie Unit 1 License Amendment 240 and Unit 2 License Amendment 191 (ADAMS Accession No. ML17219A556).

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

6. Changes have been made to be consistent with changes made to the Specification described in CTS DOCs LA01 and LA02. R1

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.6.9, SHIELD BUILDING VENTILATION SYSTEM (SBVS)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 10

ITS 3.6.10, SPRAY ADDITIVE SYSTEM - Unit 1 Only

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.6.10

3.6.2.2 The spray additive system shall be OPERABLE with:

SR 3.6.10.2

a. A spray additive tank containing a volume of between 4010 and 5000 gallons of between 28.5 and 30.5% by weight NaOH solution,

SR 3.6.10.3

and

b. ~~Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a containment spray system pump flow.~~

LA01

Applicability

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

ACTION A

With the spray additive system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6

ACTION B

hours; ~~restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours~~.

reduce pressurizer pressure to <1750 psia within 12 hours

A02

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

SR 3.6.10.1

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program by:

SR 3.6.10.2

1. Verify~~ing the contained~~ solution volume ~~in the tank, and~~

spray additive tank

is within limit of 3.6.2.2.a.

SR 3.6.10.3

2. Verify~~ing the concentration of the~~ NaOH solution ~~by chemical analysis~~.

spray additive tank

is within limit of 3.6.2.2.a

LA02

SR 3.6.10.4

c. In accordance with the Surveillance Frequency Control Program~~, during shutdown,~~ by verifying that each automatic valve in the flow path actuates to its correct position on ~~a CSAS test~~ signal.

L01

that is not locked, sealed, or otherwise secured in position,

L02

an actual or simulated actuation

L03

Applicability

* Applicable when pressurizer pressure is ≥ 1750 psia.

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.6.10.5

d. In accordance with the Surveillance Frequency Control Program by verifying a minimum sodium hydroxide (NaOH) flow rate of 10.5 gpm from the spray additive tank ~~to a drain connection immediately downstream of the tank outlet valve, and a demineralized water flow rate of 18 ± 1.5 gpm from that same drain connection to each containment spray pump~~.

LA03

~~DELETED~~

DISCUSSION OF CHANGES
ITS 3.6.10, SPRAY ADDITIVE SYSTEM – UNIT 1 ONLY

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.6.2.2 Action states in part, “restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.” ITS 3.6.10 Required Action B.2 states that if the Required Action and associated Completion Time are not met, the plant must reduce pressurizer pressure to < 1750 psia within 12 hours.” This change replaces the requirement to, “restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours,” with the requirement to reduce pressurizer pressure to < 1750 psia within 12 hours.

The purpose of the CTS 3.6.2.2 Action is to restore the system to OPERABLE status or place the unit in a condition in which the equipment is no longer required. Consistent with the Applicability of the Containment Spray System, which supports the Spray Additive System, the Applicability of CTS 3.6.2.2 only requires the Spray Additive System to be OPERABLE when pressurizer pressure is ≥ 1750 psia. The CTS 3.6.2.2 requirement to “restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours,” is inconsistent with the requirements of CTS 3.0.2 (ITS LCO 3.0.2). CTS 3.0.2 states, in part, “If the LCO is met or is no longer applicable prior to expiration of the specified time interval(s), completion of the ACTIONS is not required, unless otherwise stated.” Once pressurizer pressure is < 1750 psia, LCO 3.6.2.2 (ITS LCO 3.6.10) is not applicable and, therefore, the remaining actions are no longer required. Therefore, the requirement to restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours is no longer required to be completed.

This change is designated as administrative and has no technical impact because the requirement to “restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours,” is an action that is never applicable since the CTS (and ITS) ACTIONS require the unit to be placed in a condition where the Spray Additive System is no longer required prior to the applicable action requirements becoming limiting.

MORE RESTRICTIVE CHANGES

None

DISCUSSION OF CHANGES
ITS 3.6.10, SPRAY ADDITIVE SYSTEM – UNIT 1 ONLY

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits*) CTS 3.6.2.2.b, in part, states, "the spray additive system shall be OPERABLE with:.....b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a containment spray system pump flow." ITS 3.6.10 does not contain this level of detail. This changes the CTS by moving details concerning what constitutes an OPERABLE containment spray additive system to the Bases. |R1 |R1

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the information and is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.6.2.2.b.2 requires verification of the NaOH solution by chemical analysis. This changes the CTS by moving the procedural detail specifying the method of determining solution concentration to the Bases. |R1

The removal of these details for performing surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to perform a verification of the spray additive tank NaOH solution concentration. Also, this change is acceptable because these types of procedural detail will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.6.2.2.d states in part "to a drain connection immediately downstream of the tank outlet valve, and a demineralized water flow rate of 18 $\pm$ 1.5 gpm from that same drain connection to each containment spray pump." This changes the CTS by moving the procedural detail to the Bases.

The removal of these details for performing surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to perform a verification of minimum sodium hydroxide flow rate from the spray additive tank. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 8 – Deletion of Surveillance Requirement Shutdown Performance Requirements)* CTS 4.6.2.2.c requires verification during shutdown in accordance with the Surveillance Frequency Control Program that each automatic valve in the flow path actuates to its correct position on a CSAS signal. ITS SR 3.6.10.4 requires the same testing with no restriction as to when (i.e., during shutdown) the test can be performed. This changes the CTS by deleting the requirement to perform the Surveillance during shutdown.

The purpose of CTS 4.6.2.2.c is to ensure that each automatic valve in the Spray Additive System flow path actuates to its correct position on a CSAS. This change is acceptable because the Surveillance Requirement can be performed in other than shutdown conditions without jeopardizing safe plant operations. The control of unit conditions appropriate to perform the test is an issue for procedures and scheduling and has been determined by the NRC Staff to be unnecessary as a Technical Specification restriction. As indicated in Generic Letter 91-04, allowing this control is consistent with the vast majority of other Technical Specification Surveillances that do not dictate unit conditions for the Surveillance. This change is designated as less restrictive because the Surveillance may be performed at plant conditions other than shutdown.

L02 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.6.2.2.c requires verification that the spray additive system starts automatically on a Containment Spray Actuation Signal (CSAS). ITS SR 3.6.10.4 requires verification that each automatic valve in the flow path actuates to the correct position except for valves that are locked, sealed, or otherwise secured in the actuated position. This changes the CTS by excluding those valves that are locked, sealed, or otherwise secured in the actuated position from verification. | R1

The purpose of CTS 4.6.2.2.c is to provide assurance that if an event occurred requiring actuation of the spray additive system, then those valves and dampers requiring automatic actuation would actuate to their correct position. Those automatic valves and dampers that are locked, sealed, or otherwise secured in position are not required to actuate on a containment spray actuation signal in order to perform their safety function because they are already in the required position. Testing such valves and dampers would not provide any additional

assurance of OPERABILITY. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L03 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.6.2.2.c requires verification that the spray additive system starts automatically on a Containment Spray Actuation Signal (CSAS) test signal. ITS SR 3.6.10.4 |R1 requires verification that each automatic valve in the flow path actuates on an actual or simulated actuation signal. This changes the CTS by explicitly allowing the use of either an actual or simulated signal for the test.

The purpose of CTS 4.6.2.2.c is to ensure the spray additive system actuates upon receipt of a CSAS signal. This change is acceptable because it has been determined that the current Surveillance Requirement acceptance criteria are not the only method that can be used for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual" or "simulated" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.6 CONTAINMENT SYSTEMS

3.6.~~7~~ Spray Additive System ~~(Atmospheric and Dual)~~

3.6.2.2 LCO 3.6.~~7~~ The Spray Additive System shall be OPERABLE.

Applicability * Note APPLICABILITY: MODES 1~~,~~ and 2, ~~3, and [4].~~ MODE 3 with pressurizer pressure ≥1750 psia

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
ACTION B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND DOC A02 B.2 ~~Be in MODE 5~~.	6 hours 12 ~~84~~ hours

Reduce pressurizer pressure to < 1750 psia

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.6.2.2.a SR 3.6.~~7~~.1	Verify each spray additive manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
3.6.2.2.a 4.6.2.2.b.1	SR 3.6.~~7~~.2 10	Verify spray additive tank solution volume is ≥ ~~[816]~~ gal ~~[90%]~~ and ≤ ~~[896]~~ gal ~~[100%]~~. 4010 5000	~~[184 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1 3
3.6.2.2.a 4.6.2.2.b.2	SR 3.6.~~7~~.3 10	Verify spray additive tank ~~[~~N_2H_4~~]~~ solution concentration is ≥ ~~[33]~~% and ≤ ~~[35]~~% by weight. NaOH 28.5 30.5	~~[184 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1 3
	~~SR 3.6.7.4~~	~~[Verify each spray additive pump develops a differential pressure of [100] psid on recirculation flow.~~	~~In accordance with the INSERVICE TESTING PROGRAM]~~	3
4.6.2.2.c	SR 3.6.~~7.5~~ 10.4	Verify each spray additive automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
4.6.2.2.d	SR 3.6.~~7.6~~ 10.5	~~[~~ Verify spray additive flow ~~[~~rate~~]~~ from ~~each solution's flow path~~. a minimum; of 10.5 gpm; the spray additive tank	~~[5 years~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~ 1; 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.10, SPRAY ADDITIVE SYSTEM – UNIT 1 ONLY

1. The ISTS 3.6.7 title "Spray Additive System" includes the parenthetical expression (Dual). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Hydrogen Mixing System (ISTS 3.6.9) and the Iodine Cleanup System (ISTS 3.6.10); the PSL Unit 2 design does not include the Spray Additive System (ISTS 3.6.7). Therefore, ISTS 3.6.9 and ISTS 3.6.10 are not included in the PSL ITS. The Spray Additive System (ISTS 3.6.7) is renumbered as ITS 3.6.10, the Shield Building Exhaust Air Cleanup System (ISTS 3.6.8) is renumbered as ITS 3.6.9, the Shield Building (ISTS 3.6.11) is renumbered as ITS 3.6.7 and Vacuum Relief Valves (ISTS 3.6.12) is renumbered as ITS 3.6.8.

2. Changes are made (additions, deletions, and/or changes) to the ITS which reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. PSL Current Technical Specifications only require the Spray Additive and Containment Spray Systems to be OPERABLE in MODES 1, 2 and 3 when pressurizer pressure is ≥ 1750 psia. The Applicability of ISTS 3.6.7 (ITS 3.6.10) reflects this current licensing basis. In addition, the change made to Required Action B.2 reflects the new end state as described in Discussion of Change A02.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.~~7~~ Spray Additive System ~~(Atmospheric and Dual)~~

10

1

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere in the event of an accident such as a loss of coolant accident (LOCA).

The addition of a spray additive to the boric acid spray solution increases the pH of the spray solution and maintains the containment sump pH above 8.~~0~~ during the recirculation phase of an accident. An elevated pH is desired since it enhances the iodine removal capacity of the sprays and aids in the retention of iodine in the water in the containment sump.

.5

2

operates in conjunction with the containment spray system and

(NaOH storage)

The Spray Additive System consists of a single spray additive tank and two redundant 100% capacity trains. Each train contains a ~~chemical addition pump, an injection valve,~~ isolation valve~~s~~, a flow ~~meter~~, and ~~a flow controller~~. Upon receipt of a containment spray actuation signal (CSAS), the ~~chemical addition pumps start and the injection valves open in each redundant train. The spray additive is then injected into the Containment Spray System at the suction of the containment spray pumps at metered amounts corresponding to the individual containment spray pump discharge flow rate. The rate at which the spray additive is added is reduced when a recirculation actuation signal is generated and the Containment Spray System enters the recirculation mode of operation.~~ The pH of the containment spray solution is maintained between ~~9.0 and 10.0~~ during the injection mode and ~~between 8.0 and 9.0 in~~ the recirculation mode. Upon reaching a Low-Low level in the spray ~~chemical addition~~ tank, the ~~spray chemical addition pumps stop and the injection and~~ isolation valves close (Ref. 1).

system eductor, tank

orifice

spray nozzles

Insert 1

2

7.89 and 9.49

additive

caustic line

to isolate the tank

The Spray Additive System aids in reducing the iodine fission production inventory in the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The Spray Additive System is essential to the effective removal of airborne iodine within containment following a Design Basis Accident (DBA).

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value following the accident.

Insert 1

tank isolation valves open and stored sodium hydroxide (NaOH) is drawn into the suction of the containment spray pumps through the use of eductors. Flow rate is measured by flow orifices and is determined by eductor size and containment spray pump suction pressure.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System and is discussed in the Bases for Specification 3.6.6, "Containment Spray and Cooling Systems."

The DBA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable and that the entire spray additive tank volume is added to the remaining Containment Spray System flow path.

During a LOCA, the iodine inventory released to the containment is considered to be released instantaneously and uniformly distributed in the containment free volume. The containment volume is made up of sprayed and unsprayed regions. The sprayed region is enveloped by direct spray and mixed by the ~~dome air circulators and emergency~~ fan coolers. Mixing between the sprayed and unsprayed regions is facilitated by the ~~emergency~~ fan coolers and condensation of steam by the sprays.

containment

2

The Spray Additive System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to maintain the pH of the spray solution between ~~[9.0 and 10.0]~~ in the injection mode and ~~[8.0 and 9.0] in~~ the recirculation mode. This pH range maximizes the effectiveness of the iodine removal mechanism, without introducing conditions that may induce caustic stress corrosion cracking of mechanical components.

7.89 and 9.49

3 2

During a LOCA, one Spray Additive System train is capable of providing 100% of the required iodine removal capacity. To ensure at least one train is available in the event of the limiting single failure, both trains must be maintained in an OPERABLE status.

APPLICABILITY

In MODES 1, 2, ~~3,~~ and ~~4~~, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment.

MODE 3 ≥ 1750 psia

the Containment Spray System

In MODE 3 with pressurizer pressure <1750 psia and in MODES 4, 5,

~~In MODES 5~~ and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE ~~in MODES 5~~ and 6.

Additionally, the Containment Spray System is not required to be OPERABLE when pressurizer pressure is <1750 psia.

in MODE 3 when pressurizer pressure is <1750 psia and in MODES 4, 5,

5

BASES

ACTIONS

A.1

With the Spray Additive System inoperable, the system must be restored to OPERABLE status within 72 hours. The pH adjustment of the containment spray flow for corrosion protection and iodine removal enhancement are reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

reduce pressurizer pressure to <1750 psia

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and ~~to MODE 5~~ within ~~84~~ hours. The allowed Completion Time ~~of 6 hours is~~ reasonable, based on operating experience, to reach ~~MODE 3~~ from full power conditions in an orderly manner and without challenging plant systems. ~~The extended interval to reach MODE 5 allows additional time for restoration of the Spray Additive System and is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.~~

12
s are
the required plant conditions
5

SURVEILLANCE REQUIREMENTS

SR 3.6.~~7~~.1
10
1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned are in the correct position.

~~[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 3

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4 3

SR 3.6.~~7~~.2 10 1

volume between 4010 and 5000 gallons

a

sodium hydroxide (NaOH)

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the refueling water tank contents are normally acidic, ~~the volume of the~~ spray additive tank ~~must~~ provide s a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient ~~hydrazine (N_2H_4)~~ solution in the Spray Additive System. ~~[The 184 day Frequency is based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, such that there is a high confidence that a substantial change in level would be detected.~~ 2 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.~~7~~.3

10

This SR provides verification of the ~~N_2H_4~~ (NaOH) concentration in the spray additive tank ~~and is sufficient~~ to ensure that the spray solution being injected into containment is at the correct pH level. The concentration of ~~N_2H_4~~ (NaOH) in the spray additive tank must be determined by chemical analysis. ≥ 28.5% and ≤ 30.5% by weight

1
2

~~[The 184 day Frequency is sufficient to ensure that the concentration level of N_2H_4 in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

3

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

4
3

~~[SR 3.6.7.4~~

~~The chemical addition pump must be verified to provide the flow rate assumed in the accident analysis to the Containment Spray System. The Spray Additive System is not operated during normal operations. This prevents periodically subjecting systems, structures, and components within containment to a caustic spray solution. Therefore, this test must be performed on recirculation with the discharge flow path from each spray chemical addition pump aligned back to the spray additive tank. The differential pressure obtained by the pump on recirculation is analogous to the full spray additive flow provided to the Containment Spray System on an actual CSAS. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM and is sufficient to identify component degradation that may affect flow rate.]~~

3

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.~~7~~.5
10.4
2

This SR verifies that each automatic valve in the Spray Additive System flow path actuates to its correct position on a CSAS. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls~~. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~
3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------~~ ~~]~~
4
3

~~[~~ SR 3.6.~~7~~.6
10.5
3

To ensure that the correct pH level is established in the borated water solution provided by the Containment Spray System, the flow rate in the Spray Additive System is verified ~~once per 5 years~~. This SR provides assurance that the correct amount of ~~N_2H_4~~ will be ~~metered~~ into the flow path upon Containment Spray System initiation~~. [Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow rate.~~

periodically
6
(NaOH)
2
injected
3

~~OR~~

This SR is performed by verifying flow from the spray additive tank to a drain connection immediately downstream of the tank outlet valve, and a demineralized water flow rate of 18 ± 1.5 gpm from the same drain connection to each containment spray pump.
2 | R1

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]]~~

4
3

REFERENCES 1. FSAR, Section ~~[~~ ~~]~~.

6.2
U
3
2

JUSTIFICATION FOR DEVIATIONS
ITS 3.6.10 BASES, SPRAY ADDITIVE SYSTEM – UNIT 1 ONLY

1. The ISTS 3.6.7 title "Spray Additive System" includes the parenthetical expression (Dual). This identifying information is not included in the PSL ITS. This information is provided in the NUREG-1432, Rev. 5.0 to assist in identifying the appropriate Specifications to be used as a model for a plant-specific ITS conversion but serves no purpose in a plant-specific implementation. In addition, PSL design does not include the Hydrogen Mixing System (ISTS 3.6.9) and the Iodine Cleanup System (ISTS 3.6.10); the PSL Unit 2 design does not include the Spray Additive System (ISTS 3.6.7). Therefore, ISTS 3.6.9 and ISTS 3.6.10 are not included in the PSL ITS. The Spray Additive System (ISTS 3.6.7) is renumbered as ITS 3.6.10, the Shield Building Exhaust Air Cleanup System (ISTS 3.6.8) is renumbered as ITS 3.6.9, the Shield Building (ISTS 3.6.11) is renumbered as ITS 3.6.7 and Vacuum Relief Valves (ISTS 3.6.12) is renumbered as ITS 3.6.8.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is changed to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Concomitant Bases changes made to reflect changes to the ISTS Specification regarding change to Applicability and ACTIONS to specify the end state as MODE 3 with a pressurizer pressure of 1750 psia.

6. Text changed to reflect that the Frequency is performed periodically. This change is consistent with similar ISTS Bases changes as a result of incorporation of TSTF-425, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b," into the ISTS.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.6.10, SPRAY ADDITIVE SYSTEM – UNIT 1 ONLY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 11

ISTS Not Adopted

ISTS 3.6.7, SPRAY ADDITIVE SYSTEM (UNIT 2 ONLY)

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.6 CONTAINMENT SYSTEMS

3.6.7 Spray Additive System (Atmospheric and Dual)

LCO 3.6.7 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and [4].

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.7.1 Verify each spray additive manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	[31 days OR In accordance with the Surveillance Frequency Control Program]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.7.2	Verify spray additive tank solution volume is ≥ [816] gal [90%] and ≤ [896] gal [100%].	[184 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.6.7.3	Verify spray additive tank [N_2H_4] solution concentration is ≥ [33]% and ≤ [35]% by weight.	[184 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.6.7.4	[Verify each spray additive pump develops a differential pressure of [100] psid on recirculation flow.	In accordance with the INSERVICE TESTING PROGRAM]
SR 3.6.7.5	Verify each spray additive automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	[[18] months OR In accordance with the Surveillance Frequency Control Program]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.7.6	[Verify spray additive flow [rate] from each solution's flow path.	[5 years OR In accordance with the Surveillance Frequency Control Program]]

JUSTIFICATION FOR DEVIATIONS
ISTS 3.6.7, SPRAY ADDITIVE SYSTEM

1. ISTS 3.6.7, “Spray Additive System” is not being adopted for PSL Unit 2. The PSL Unit 2 technical specification requirements for the Iodine Removal System were eliminated with “Issuance of Amendment No. 201 Regarding Technical Specification Changes to Eliminate the Requirements of the Iodine Removal System,” dated November 20, 2019 (ADAMS Accession No. ML19248C238). Therefore, ISTS 3.6.7 is not included in the PSL Unit 2 ITS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System (Atmospheric and Dual)

BASES

BACKGROUND The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere in the event of an accident such as a loss of coolant accident (LOCA).

The addition of a spray additive to the boric acid spray solution increases the pH of the spray solution and maintains the containment sump pH above 8.0 during the recirculation phase of an accident. An elevated pH is desired since it enhances the iodine removal capacity of the sprays and aids in the retention of iodine in the water in the containment sump.

The Spray Additive System consists of a single spray additive tank and two redundant 100% capacity trains. Each train contains a chemical addition pump, an injection valve, isolation valves, a flow meter, and a flow controller. Upon receipt of a containment spray actuation signal (CSAS), the chemical addition pumps start and the injection valves open in each redundant train. The spray additive is then injected into the Containment Spray System at the suction of the containment spray pumps at metered amounts corresponding to the individual containment spray pump discharge flow rate. The rate at which the spray additive is added is reduced when a recirculation actuation signal is generated and the Containment Spray System enters the recirculation mode of operation. The pH of the containment spray solution is maintained between 9.0 and 10.0 during the injection mode and between 8.0 and 9.0 in the recirculation mode. Upon reaching a Low-Low level in the spray chemical addition tank, the spray chemical addition pumps stop and the injection and isolation valves close (Ref. 1).

The Spray Additive System aids in reducing the iodine fission production inventory in the containment atmosphere.

APPLICABLE SAFETY ANALYSES The Spray Additive System is essential to the effective removal of airborne iodine within containment following a Design Basis Accident (DBA).

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value following the accident.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System and is discussed in the Bases for Specification 3.6.6, "Containment Spray and Cooling Systems."

The DBA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable and that the entire spray additive tank volume is added to the remaining Containment Spray System flow path.

During a LOCA, the iodine inventory released to the containment is considered to be released instantaneously and uniformly distributed in the containment free volume. The containment volume is made up of sprayed and unsprayed regions. The sprayed region is enveloped by direct spray and mixed by the dome air circulators and emergency fan coolers. Mixing between the sprayed and unsprayed regions is facilitated by the emergency fan coolers and condensation of steam by the sprays.

The Spray Additive System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to maintain the pH of the spray solution between [9.0 and 10.0] in the injection mode and [8.0 and 9.0] in the recirculation mode. This pH range maximizes the effectiveness of the iodine removal mechanism, without introducing conditions that may induce caustic stress corrosion cracking of mechanical components.

During a LOCA, one Spray Additive System train is capable of providing 100% of the required iodine removal capacity. To ensure at least one train is available in the event of the limiting single failure, both trains must be maintained in an OPERABLE status.

APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODES 5 and 6.

BASES

ACTIONS A.1

With the Spray Additive System inoperable, the system must be restored to OPERABLE status within 72 hours. The pH adjustment of the containment spray flow for corrosion protection and iodine removal enhancement are reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for restoration of the Spray Additive System and is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned are in the correct position.

[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the refueling water tank contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient hydrazine (N_2H_4) solution in the Spray Additive System. [The 184 day Frequency is based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, such that there is a high confidence that a substantial change in level would be detected.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.3

This SR provides verification of the N_2H_4 concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The concentration of N_2H_4 in the spray additive tank must be determined by chemical analysis. [The 184 day Frequency is sufficient to ensure that the concentration level of N_2H_4 in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

[SR 3.6.7.4

The chemical addition pump must be verified to provide the flow rate assumed in the accident analysis to the Containment Spray System. The Spray Additive System is not operated during normal operations. This prevents periodically subjecting systems, structures, and components within containment to a caustic spray solution. Therefore, this test must be performed on recirculation with the discharge flow path from each spray chemical addition pump aligned back to the spray additive tank. The differential pressure obtained by the pump on recirculation is analogous to the full spray additive flow provided to the Containment Spray System on an actual CSAS. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM and is sufficient to identify component degradation that may affect flow rate.]

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.5

This SR verifies that each automatic valve in the Spray Additive System flow path actuates to its correct position on a CSAS. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

[SR 3.6.7.6

To ensure that the correct pH level is established in the borated water solution provided by the Containment Spray System, the flow rate in the Spray Additive System is verified once per 5 years. This SR provides assurance that the correct amount of N_2H_4 will be metered into the flow path upon Containment Spray System initiation. [Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow rate.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]]

REFERENCES 1. FSAR, Section [ ].

JUSTIFICATION FOR DEVIATIONS
ISTS 3.6.7 BASES, SPRAY ADDITIVE SYSTEM

1. ISTS 3.6.7, “Spray Additive System” is not being adopted for PSL Unit 2. The PSL Unit 2 technical specification requirements for the Iodine Removal System were eliminated with “Issuance of Amendment No. 201 Regarding Technical Specification Changes to Eliminate the Requirements of the Iodine Removal System,” dated November 20, 2019 (ADAMS Accession No. ML19248C238). Therefore, the Bases associated with ISTS 3.6.7 is not included in the PSL Unit 2 ITS Bases.

ISTS 3.6.9, HYDROGEN MIXING SYSTEM (HMS) (ATMOSPHERIC AND DUAL)

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.6 CONTAINMENT SYSTEMS

3.6.9 Hydrogen Mixing System (HMS) (Atmospheric and Dual)

LCO 3.6.9 [Two] HMS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One HMS train inoperable.	A.1 Restore HMS train to OPERABLE status.	30 days
B. [Two HMS trains inoperable.	B.1 Verify by administrative means that the hydrogen control function is maintained. AND B.2 Restore one HMS train to OPERABLE status.	1 hour AND Once every 12 hours thereafter 7 days]
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.9.1	Operate each HMS train for ≥ 15 minutes.	[92 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.6.9.2	Verify each HMS train flow rate on slow speed is ≥ [37,000] cfm.	[[18] months OR In accordance with the Surveillance Frequency Control Program]
SR 3.6.9.3	Verify each HMS train starts on an actual or simulated actuation signal.	[[18] months OR In accordance with the Surveillance Frequency Control Program]

JUSTIFICATION FOR DEVIATIONS
ISTS 3.6.9, HYDROGEN MIXING SYSTEM (HMS) (ATMOSPHERIC AND DUAL)

1. ISTS 3.6.9, “Hydrogen Mixing System (HMS) (Atmospheric and Dual)” is not being adopted because St. Lucie Plant (PSL) design does not include the HMS. The hydrogen mixing function is performed by ITS 3.6.6, “Containment Spray and Cooling System.” Therefore, ISTS 3.6.9 is not included in the PSL Unit 1 and 2 ITS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Hydrogen Mixing System (HMS) (Atmospheric and Dual)

BASES

BACKGROUND

The HMS reduces the potential for breach of containment due to a hydrogen oxygen reaction by providing a uniformly mixed post accident containment atmosphere, thereby minimizing the potential for local hydrogen burns due to a local pocket of hydrogen above the flammable concentration and giving the operator the capability of preventing the occurrence of a bulk hydrogen burn inside containment per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and 10 CFR 50, GDC 41, "Containment Atmosphere Cleanup" (Ref. 2).

The post accident HMS is an Engineered Safety Feature and is designed to withstand a loss of coolant accident (LOCA) without loss of function. The system has two independent trains, each of which consists of two dome air circulation fans, motors, and controls. Each train is sized for [37,000] cfm. The two trains are initiated automatically on a containment cooling actuation signal (CCAS) or can be manually started from the control room. Each train is powered from a separate emergency power supply. Since each train can provide 100% of the mixing requirements, the system will provide its design function with a limiting single active failure.

The HMS accelerates the air mixing process between the upper dome space of the containment atmosphere during LOCA operations. It also prevents any hot spot air pockets during the containment cooling mode and avoids any hydrogen concentration in pocket areas.

Hydrogen mixing within the containment is accomplished by the Containment Spray System, the containment emergency fan coolers, and the containment internal structure design, which permits convective mixing and prevents entrapment. The HMS, operating in conjunction with the Containment Spray System and the emergency fan coolers, prevents localized accumulations of hydrogen from exceeding the flammability limit of 4.1 volume percent (v/o).

APPLICABLE SAFETY ANALYSES

The HMS mixes the containment atmosphere to provide a uniform hydrogen concentration.

Hydrogen may accumulate in containment following a LOCA as a result of:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. A metal-steam reaction between the zirconium fuel rod cladding and the reactor coolant,

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump,

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space), or

d. Corrosion of metals exposed to Containment Spray System and Emergency Core Cooling Systems solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

The HMS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two HMS trains must be OPERABLE, with power to each from an independent, safety related power supply. Each train typically consists of two fans with their own motors and controls and is automatically initiated by a CCAS.

Operation with at least one HMS train provides the mixing necessary to ensure uniform hydrogen concentration throughout containment.

APPLICABILITY

In MODES 1 and 2, the two HMS trains ensure the capability to prevent localized hydrogen concentrations above the flammability limit of 4.1 v/o in containment, assuming a worst case single active failure.

In MODE 3 or 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the HMS is low. Therefore, the HMS is not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA or main steam line break are low due to the pressure and temperature limitations of these MODES. Therefore, the HMS is not required in these MODES.

BASES

ACTIONS A.1

With one HMS train inoperable, the inoperable train must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on the availability of the other HMS train, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit, and the availability of the Containment Spray System and Hydrogen Purge System.

B.1 and B.2

REVIEWER'S NOTE

This Condition is only allowed for units with an alternate hydrogen control system acceptable to the technical staff.

With two HMS inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by [the containment Hydrogen Purge System/Hydrogen Ignitor System/HMS/Containment Air Dilution System/Containment Inerting System]. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist.

REVIEWER'S NOTE

The following is to be used if a non-Technical Specification alternate hydrogen control function is used to justify this Condition. In addition, the alternate hydrogen control system capability must be verified every 12 hours thereafter to ensure its continued availability.

[Both] the [initial] verification [and all subsequent verifications] may be performed as an administrative check, by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two HMS trains inoperable for up to 7 days. Seven days is a reasonable time to allow two HMS trains to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

BASES

ACTIONS (continued)

C.1

If an inoperable HMS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

Operating each HMS train for ≥ 15 minutes ensures that the train is OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. [The 92 day Frequency is consistent with INSERVICE TESTING PROGRAM Surveillance Frequencies, operating experience, the known reliability of the fan motors and controls, and the two train redundancy available.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

SR 3.6.9.2

Verifying that each HMS train flow rate on slow speed is ≥ [37,000] cfm ensures that each train is capable of maintaining localized hydrogen concentrations below the flammability limit. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

SR 3.6.9.3

This SR ensures that the HMS responds properly to a CCAS. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

REFERENCES 1. 10 CFR 50.44.

2. 10 CFR 50, Appendix A, GDC 41.

3. Regulatory Guide 1.7, Revision [1].

JUSTIFICATION FOR DEVIATIONS
ISTS 3.6.9 BASES, HYDROGEN MIXING SYSTEM (HMS) (ATMOSPHERIC AND DUAL)

1. ISTS 3.6.9, “Hydrogen Mixing System (HMS) (Atmospheric and Dual)” is not being adopted because St. Lucie Plant (PSL) design does not include the HMS. The hydrogen mixing function is performed by ITS 3.6.6, “Containment Spray and Cooling System.” Therefore, the Bases associated with ISTS 3.6.9 is not included in the PSL Unit 1 and Unit 2 ITS Bases.

ISTS 3.6.10, IODINE CLEANUP SYSTEM (ICS) (ATMOSPHERIC AND DUAL)

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.6 CONTAINMENT SYSTEMS

3.6.10 Iodine Cleanup System (ICS) (Atmospheric and Dual)

LCO 3.6.10 Two ICS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ICS train inoperable.	A.1 Restore ICS train to OPERABLE status.	7 days
B. ---------- NOTE ---------- Not applicable when second ICS train intentionally made inoperable. ---------- Two ICS trains inoperable.	B.1 Verify at least one train of containment spray is OPERABLE. AND B.2 Restore at least one ICS train to OPERABLE status.	1 hour 24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.10.1	Operate each ICS train for ≥ 15 continuous minutes [with heaters operating].	[31 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.6.10.2	Perform required ICS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.10.3	Verify each ICS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	[[18] months OR In accordance with the Surveillance Frequency Control Program]
SR 3.6.10.4	[Verify each ICS filter bypass damper can be opened, except for dampers that are locked, sealed, or otherwise secured in the open position.	[[18] months OR In accordance with the Surveillance Frequency Control Program]]

JUSTIFICATION FOR DEVIATIONS
ISTS 3.6.10, IODINE CLEANUP SYSTEM (ICS) (ATMOSPERIC AND DUAL)

1. ISTS 3.6.10, “Iodine Cleanup System (ICS) (Atmospheric and Dual)” is not being adopted. The iodine removal function is performed by ITS 3.6.6, “Containment Spray and Cooling System” for PSL Unit 1. The PSL Unit 2 technical specification requirements for the Iodine Removal System were eliminated with “Issuance of Amendment No. 201 Regarding Technical Specification Changes to Eliminate the Requirements of the Iodine Removal System,” dated November 20, 2019 (ADAMS Accession No. ML19248C238). Therefore, ISTS 3.6.10 is not included in the PSL Unit 1 and 2 ITS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Iodine Cleanup System (ICS) (Atmospheric and Dual)

BASES

BACKGROUND The ICS is provided per GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1), to reduce the concentration of fission products released to the containment atmosphere following a postulated accident. The ICS would function together with the Containment Spray and Cooling systems following a Design Basis Accident (DBA) to reduce the potential release of radioactive material, principally iodine, from the containment to the environment.

The ICS consists of two 100% capacity separate, independent, and redundant trains. Each train includes a heater, [cooling coils,] a prefilter, a moisture separator, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of radioiodines, and a fan. Ductwork, valves and/or dampers, and instrumentation also form part of the system. The moisture separators function to reduce the moisture content of the airstream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates filtered recirculation of the containment atmosphere following receipt of a containment isolation actuation signal. The system design is described in Reference 2.

The primary purpose of the heaters is to ensure that the relative humidity of the airstream entering the charcoal adsorbers is maintained below 70%, which is consistent with the assigned iodine and iodide removal efficiencies as per Regulatory Guide 1.52 (Ref. 3).

The moisture separator is included for moisture (free water) removal from the gas stream. Heaters are used to heat the gas stream, which lowers the relative humidity. Both the moisture separator and heater are important to the effectiveness of the charcoal adsorbers.

Two ICS trains are provided to meet the requirement for separation, independence, and redundancy. Each ICS train is powered from a separate Engineered Safety Features bus and is provided with a separate power panel and control panel. [Service water is required to supply cooling water to the cooling coils.]

BASES

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive iodine within containment are a loss of coolant accident (LOCA), a main steam line break (MSLB), or a control element assembly (CEA) ejection accident. In the analysis for each of these accidents, it is assumed that adequate containment leak tightness is intact at event initiation to limit potential leakage to the environment. Additionally, it is assumed that the amount of radioactive iodine release is limited by reducing the iodine concentration in the containment atmosphere.

The ICS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 4) assumes that only one train of the ICS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive iodine provided by the remaining one train of this filtration system.

The ICS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two separate, independent, and redundant trains of the ICS are required to ensure that at least one is available, assuming a single failure coincident with a loss of offsite power.

APPLICABILITY

In MODES 1, 2, 3, and 4, iodine is a fission product that can be released from the fuel to the reactor coolant as a result of a DBA. The DBAs that can cause a failure of the fuel cladding are a LOCA, MSLB, and CEA ejection accident. Because these accidents are considered credible accidents in MODES 1, 2, 3, and 4, the ICS must be operable in these MODES to ensure the reduction in iodine concentration assumed in the accident analysis.

In MODES 5 and 6, the probability and consequences of a LOCA are low due to the pressure and temperature limitations of these MODES. The ICS is not required in these MODES to remove iodine from the containment atmosphere.

ACTIONS

A.1

With one ICS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as:

a. The availability of the OPERABLE redundant ICS train,

BASES

ACTIONS (continued)

b. The fact that, even with no ICS train in operation, almost the same amount of iodine would be removed from the containment atmosphere through absorption by the Containment Spray System, and

c. The fact that the Completion Time is adequate to make most repairs.

B.1 and B.2

If two ICS trains are inoperable, at least one ICS train must be returned to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second ICS train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. In addition, at least one train of containment spray must be verified to be OPERABLE within 1 hour. In the event of an accident, containment spray reduces the potential radioactive release from the containment, which reduces the consequences of the inoperable ICS trains. The Completion Time is based on Reference 5 which demonstrated that the 24 hour Completion Time is acceptable based on the infrequent use of the Required Actions and the small incremental effect on plant risk.

C.1 and C.2

If the ICS train(s) cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

Operating each ICS train for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. [The 31 day Frequency was developed considering the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System independent of the ICS.

BASES

SURVEILLANCE REQUIREMENTS (continued)

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

SR 3.6.10.2

This SR verifies that the required ICS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.10.3

The automatic startup test verifies that both trains of equipment start upon receipt of an actual or simulated test signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the

SURVEILLANCE REQUIREMENTS (continued)

Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the Frequency was developed considering that the system equipment OPERABILITY is demonstrated on a 31 day Frequency by SR 3.6.10.1.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

[SR 3.6.10.4

The ICS filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal operation and must reposition for accident operation to draw air through the filters. The SR excludes automatic dampers that are locked, sealed, or otherwise secured in the open position. The SR does not apply to dampers that are locked, sealed, or otherwise secured in the open position since the affected dampers were verified to be in the open position prior to being locked, sealed, or otherwise secured. Placing an automatic damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the damper to be closed to support the accident analysis. Restoration of an automatic damper to the closed position requires verification that the SR has been met within its required Frequency. [The [18] month Frequency is considered to be acceptable based on the damper reliability and design, the mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the Surveillance when performed at the [18] month Frequency.

OR

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------REVIEWER'S NOTE----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]]

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41, GDC 42, and GDC 43.

2. FSAR, Section [ ].

3. Regulatory Guide 1.52, Revision [2].

4. FSAR, Section [ ].

5. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

JUSTIFICATION FOR DEVIATIONS
ISTS 3.6.10 BASES, IODINE CLEANUP SYSTEM (ICS) (ATMOSPERIC AND DUAL)

1. ISTS 3.6.10, “Iodine Cleanup System (ICS) (Atmospheric and Dual)” is not being adopted. The iodine removal function is performed by ITS 3.6.6, “Containment Spray and Cooling System” for PSL Unit 1. The PSL Unit 2 technical specification requirements for the Iodine Removal System were eliminated with “Issuance of Amendment No. 201 Regarding Technical Specification Changes to Eliminate the Requirements of the Iodine Removal System,” dated November 20, 2019 (ADAMS Accession No. ML19248C238). Therefore, the Bases associated with ISTS 3.6.10 is not included in the PSL Unit 1 and Unit 2 ITS Bases.

ISTS 3.6.13, CONTAINMENT SUMP

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.6 CONTAINMENT SYSTEMS

3.6.13 Containment Sump

LCO 3.6.13 [The][Two] containment sump[s] shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. [One or more] containment sump[s] inoperable due to containment accident generated and transported debris exceeding the analyzed limits.	A.1 Initiate action to mitigate containment accident generated and transported debris. AND A.2 Perform SR 3.4.13.1. AND A.3 Restore the containment sump[s] to OPERABLE status.	Immediately Once per 24 hours 90 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. [One or more] containment sump[s] inoperable for reasons other than Condition A.	B.1 ----------NOTES---------- 1. Enter applicable Conditions and Required Actions of LCO 3.5.2, "ECCS - Operating," and LCO 3.5.3, "ECCS - Shutdown," for emergency core cooling trains made inoperable by the containment sump[s]. 2. Enter applicable Conditions and Required Actions of LCO 3.6.6, "Containment Spray and Cooling Systems," for containment spray trains made inoperable by the containment sump[s]. ---------- Restore the containment sump[s] to OPERABLE status.	[72 hours] [OR In accordance with the Risk Informed Completion Time Program]
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.13.1	Verify, by visual inspection, the containment sump[s] does not show structural damage, abnormal corrosion, or debris blockage.	[[18] months OR In accordance with the Surveillance Frequency Control Program]

JUSTIFICATION FOR DEVIATIONS
ISTS 3.6.13, CONTAINMENT SUMP

1. ISTS 3.6.13, “Containment Sump” is not being adopted. The Containment Sump function is performed by ITS 3.5.2, “ECCS – Operating”, ITS 3.5.3, “ECCS – Shutdown” and ITS 3.6.6, “Containment Spray and Cooling System” for PSL Unit 1 and Unit 2. Therefore, ISTS 3.6.13 is not included in the PSL Unit 1 and 2 ITS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.13 Containment Sump

BASES

BACKGROUND ---------- REVIEWER'S NOTE ----------
Some plant designs have more than one containment sump. LCO 3.6.13 provides the option of specifying one or two containment sumps. The Bases specify when to discuss differences between one or more sump designs by providing bracketed information. However, for clarity, all references to the containment sump are not provided with plural bracketed alternatives. Licensees adopting this Bases section for plants with two or more containment sumps should make the appropriate changes to the Bases to reflect the plant design.

The containment sump provides a borated water source to support recirculation of coolant from the containment sump for residual heat removal, emergency core cooling, containment cooling, and [containment atmosphere cleanup] during accident conditions.

The containment sump supplies both trains of the Emergency Core Cooling System (ECCS) and the Containment Spray System (CSS) during any accident that requires recirculation of coolant from the containment sump. The recirculation mode is initiated when the pump suction is transferred to the containment sump on low Refueling Water Tank (RWT) level, which ensures the containment sump has enough water to supply the net positive suction head to the ECCS and CSS pumps. [The use of a single containment sump to supply both trains of the ECCS and CSS is acceptable since the containment sump is a passive component, and passive failures are not required to be assumed to occur coincident with Design Basis Events.][Describe the design of plants with two or more containment sumps.]

The containment sump contains strainers to limit the quantity of the debris materials from entering the sump suction piping. Debris accumulation on the strainers can lead to undesirable hydraulic effects including air ingestion through vortexing or deaeration, and reduced net positive suction head (NPSH) at pump suction piping.

BASES

BACKGROUND (continued)

While the majority of debris accumulates on the strainers, some fraction penetrates the strainers and is transported to downstream components in the ECCS, CSS, and the Reactor Coolant System (RCS). Debris that penetrates the strainer can result in wear to the downstream components, blockages, or reduced heat transfer across the fuel cladding. Excessive debris in the containment sump water source could result in insufficient recirculation of coolant during the accident, or insufficient heat removal from the core during the accident.

APPLICABLE SAFETY ANALYSIS

During all accidents that require recirculation, the containment sump provides a source of borated water to the ECCS and CSS pumps. As such, it supports residual heat removal, emergency core cooling, containment cooling, and [containment atmosphere cleanup] during an accident. It also provides a source of negative reactivity (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating," B 3.5.3, "ECCS - Shutdown," and B 3.6.6, "Containment Spray and Cooling Systems."

FSAR Section X.XX (Ref. 2) describes evaluations that confirm long-term core cooling is assured following any accident that requires recirculation from the containment sump.

The containment sump satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

---------- REVIEWER'S NOTE ----------
The containment sump description should include all design features credited in the containment debris analysis.

[The][Two] containment sump is required to ensure a source of borated water to support ECCS and CSS OPERABILITY. A containment sump consists of the containment drainage flow paths, [design features upstream of the containment sump that are credited in the containment debris analysis,] the containment sump strainers, the pump suction trash racks, and the inlet to the ECCS and CSS piping. An OPERABLE containment sump has no structural damage or abnormal corrosion that could prevent recirculation of coolant and will not be restricted by containment accident generated and transported debris.

BASES

LCO (continued)

Containment accident generated and transported debris consists of the following:

a. Accident generated debris sources - Insulation, coatings, and other materials which are damaged by the high-energy line break (HELB) and transported to the containment sump. This includes materials within the HELB zone of influence and other materials (e.g., unqualified coatings) that fail due to the post-accident containment environment following the accident;

b. Latent debris sources – Pre-existing dirt, dust, paint chips, fines or shards of insulation, and other materials inside containment that do not have to be damaged by the HELB to be transported to the containment sump; and

c. Chemical product debris sources – Aluminum, zinc, carbon steel, copper, and non-metallic materials such as paints, thermal insulation, and concrete that are susceptible to chemical reactions within the post-accident containment environment leading to corrosion products that are generated within the containment sump pool or are generated within containment and transported to the containment sump.

Containment debris limits are defined in FSAR Section X.XX (Ref. 2).

APPLICABILITY In MODES 1, 2, 3, and 4, containment sump OPERABILITY requirements are dictated by the ECCS and CSS OPERABILITY requirements. Since both the ECCS and the CSS must be OPERABLE in MODES 1, 2, 3, and 4, the containment sump must also be OPERABLE to support their operation.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the containment sump is not required to be OPERABLE in MODES 5 or 6.

ACTIONS A.1, A.2, and A.3

Condition A is applicable when there is a condition which results in containment accident generated and transported debris exceeding the analyzed limits. Containment debris limits are defined in FSAR Section X.XX (Ref. 2).

BASES

ACTIONS (continued)

Immediate action must be initiated to mitigate the condition. Examples of mitigating actions are:

- Removing the debris source from containment or preventing the debris from being transported to the containment sump;
- Evaluating the debris source against the assumptions in the analysis;
- Deferring maintenance that would affect availability of the affected systems and other LOCA mitigating equipment;
- Deferring maintenance that would affect availability of primary defense-in-depth systems, such as containment coolers;
- Briefing operators on LOCA debris management actions; or
- Applying an alternative method to establish new limits.

While in this condition, the RCS water inventory balance, SR 3.4.13.1, must be performed at an increased Frequency of once per 24 hours. An unexpected increase in RCS leakage could be indicative of an increased potential for an RCS pipe break, which could result in debris being generated and transported to the containment sump. The more frequent monitoring allows operators to act in a timely fashion to minimize the potential for an RCS pipe break while the containment sump is inoperable.

[For the purposes of applying LCO 3.0.6 and the Safety Function Determination Program while in Condition A, the [two] containment sumps are considered a single support system for all ECCS and CSS trains because containment accident generated and transported debris issues that would render one sump inoperable would render all of the sumps inoperable.]

The inoperable containment sump must be restored to OPERABLE status in 90 days. A 90-day Completion Time is reasonable for emergent conditions that involve debris in excess of the analyzed limits that could be generated and transported to the containment sump under accident conditions. The likelihood of an initiating event in the 90-day Completion Time is very small and there is margin the associated analyses. The mitigating actions of Required Action A.1 provide additional assurance that the effects of debris in excess of the analyzed limits will be mitigated during the Completion Time.

BASES

ACTIONS (continued)

B.1

---------- Reviewer's Note ----------
The Required Action B.1 Completion Time should be the same as the LCO 3.6.6 Completion Time for a single inoperable CSS train.

When the containment sump is inoperable for reasons other than Condition A, such as blockage, structural damage, or abnormal corrosion that could prevent recirculation of coolant, it must be restored to OPERABLE status within [72 hours]. The [72 hour] Completion Time takes into account the reasonable time for repairs, and low probability of an accident that requires the containment sump occurring during this period. [Alternately, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]

Required Action B.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.5.2, "ECCS - Operating," and LCO 3.5.3, "ECCS - Shutdown," should be entered if an inoperable containment sump results in an inoperable ECCS train. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.6.6, "Containment Spray and Cooling Systems," should be entered if an inoperable containment sump results in an inoperable CSS train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

C.1 and C.2

If the containment sump cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.13.1

Periodic inspections are performed to verify the containment sump does not show current or potential debris blockage, structural damage, or abnormal corrosion to ensure the operability and structural integrity of the containment sump (Ref. 1).

[The 18 month Frequency is based on the need to perform this Surveillance during a refueling outage, because of the need to enter containment. This Frequency is sufficient to detect any indication of structural damage, abnormal corrosion, or debris blockage of the containment sump.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

REFERENCES

1. FSAR, Chapter [6] and Chapter [15].

2. FSAR, Section [X.XX], [Sump Debris Evaluation].

JUSTIFICATION FOR DEVIATIONS
ISTS 3.6.13 BASES, CONTAINMENT SUMP

1. ISTS 3.6.13, “Containment Sump” is not being adopted. The containment sump function is performed by ITS 3.5.2, “ECCS – Operating”, ITS 3.5.3, “ECCS – Shutdown” and ITS 3.6.6, “Containment Spray and Cooling System” for PSL Unit 1 and Unit 2. Therefore, the Bases associated with ISTS 3.6.10 is not included in the PSL Unit 1 and Unit 2 ITS Bases.

ENCLOSURE 2

VOLUME 12

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.7
PLANT SYSTEMS

Revision 1

LIST OF ATTACHMENTS

1. 3.7.1, Main Steam Safety Valves (MSSVs)
2. 3.7.2, Main Steam Isolation Valves (MSIVs)
3. 3.7.3, Main Feedwater Isolation Valves (MFIVs)
4. 3.7.4, Atmospheric Dump Valves (ADVs)
5. 3.7.5, Auxiliary Feedwater (AFW) System
6. 3.7.6, Condensate Storage Tank (CST)
7. 3.7.7, Component Cooling Water (CCW) System
8. 3.7.8, Intake Cooling Water System (ICW)
9. 3.7.9, Ultimate Heat Sink (UHS)
10. 3.7.10, Control Room Emergency Ventilation System (CREVS)
11. 3.7.11, Control Room Air Conditioning System (CRACS)
12. 3.7.12, Emergency Core Cooling System (ECCS) Area Ventilation System
13. 3.7.13, Fuel Storage Pool Water Level
14. 3.7.14, Fuel Storage Pool Boron Concentration
15. 3.7.15, Spent Fuel Pool Storage
16. 3.7.16, Secondary Specific Activity
17. 3.7.17, Fuel Pool Area Ventilation System - Unit 1
18. Relocated/Deleted Current Technical Specifications
19. ISTS Not Adopted

ATTACHMENT 1

ITS 3.7.1, Main Steam Safety Valves (MSSVs)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3/4.7 PLANT SYSTEMS

3.4.7.1 TURBINE CYCLE

MAIN STEAM (MSSVs)

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

Eight MSSVs per steam generator

LCO 3.7.1 / SR 3.7.1.1

3.7.1.1 ~~All main steam line code safety valves~~ shall be OPERABLE with lift settings as specified in Table 4.7-1. A02

Applicability

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

A03

Add proposed ACTIONS Note

and

Add proposed Required Action A.1

A04

required MSSVs

O

a. ~~With both reactor coolant loops and associated steam generators in operation and with~~ one or more ~~main steam line code safety valves~~ inoperable, operation in MODES 1, 2 and 3 may proceed provided that within 4 hours, ~~either the inoperable valve is restored to OPERABLE status or~~ the Power Level-High trip setpoint is reduced per Table 3.7-1, ~~otherwise~~, be in at least ~~HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following 30~~ hours.

Action A

Action B

A05

A06

MODE 3

within 36 hours

L01

MODE 4

12

Required Action and associated Completion Time not met or one or more steam generators with less than five MSSVs OPERABLE

SURVEILLANCE REQUIREMENTS

Add proposed SR Note

L02

SR 3.7.1.1

4.7.1.1 Verify each ~~main steam line code safety valve is OPERABLE~~ in accordance with the INSERVICE TESTING PROGRAM. Following testing, ~~as-left~~ lift settings shall be within +/- 1% ~~of 1000 psia for valves 8201 through 8208, and within +/- 1% of 1040 psia for valves 8209 through 8216 specified in Table 4.7-1~~.

LA01

required MSSV lift setpoint per Table 3.7.1-2

Table 3.7.1-1

TABLE 3.7-1

VARIABLE

VERSUS

MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT ~~WITH IN~~OPERABLE

MAIN

STEAM ~~LINE~~ SAFETY VALVES ~~DURING OPERATION WITH BOTH STEAM GENERATORS~~

Minimum MSSVs per ~~Maximum~~ Number of ~~Inoperable Safety Valves on Any Operating~~ Steam Generator	Required Operable	Variable Maximum Allowable Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
~~1~~	7	88.5
~~2~~	6	79.8
~~3~~	5	66.5

A04

3.7.1-2

TABLE ~~4.7-1~~

Main Steam Safety Valve Lift Settings

~~STEAM LINE SAFETY VALVES PER LOOP~~

Table 3.7.1-2

	VALVE NUMBER		LIFT SETTING*
	Steam Generator 1A ~~Header A~~	Steam Generator 1B ~~Header B~~	
Bank 1	a. 8201	8205	~~≥ 955.3 psig and ≤ 1015.3 psig~~
	b. 8202	8206	~~≥ 955.3 psig and ≤ 1015.3 psig~~
	c. 8203	8207	~~≥ 955.3 psig and ≤ 1015.3 psig~~
	d. 8204	8208	~~≥ 955.3 psig and ≤ 1015.3 psig~~
Bank 2	e. 8209	8213	~~≥ 994.1 psig and ≤ 1046.1 psig~~
	f. 8210	8214	~~≥ 994.1 psig and ≤ 1046.1 psig~~
	g. 8211	8215	~~≥ 994.1 psig and ≤ 1046.1 psig~~
	h. 8212	8216	~~≥ 994.1 psig and ≤ 1046.1 psig~~
	V	V	

985 psig

1025 psig

A07

---

* +/-3% for ~~valves a through d~~ and +2%/-3% for ~~valves e through h~~

Bank 1 Bank 2

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

MAIN STEAM (MSSVs)

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

Eight MSSVs per steam generator

LCO 3.7.1 / SR 3.7.1.1

3.7.1.1 ~~All main steam line code safety valves~~ shall be OPERABLE with lift settings as shown in Table 3.7-2.

A02

Applicability

APPLICABILITY: MODES 1, 2 and 3.

A03

ACTION:

Add proposed ACTIONS Note

and

Add proposed Required Action A.1

A04

required MSSVs

O

a. ~~With both reactor coolant loops and associated steam generators in operation and with~~ one or more ~~main steam line code safety valves~~ inoperable, operation in MODES 1, 2 and 3 may proceed provided that, within 4 hours, ~~either the inoperable valve is restored to OPERABLE status or~~ the Power Level-High trip setpoint is reduced per Table 3.7-1, ~~otherwise~~, be in at least ~~HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following 30~~ hours.

Action A

Action B

A05

A06

MODE 3

12

MODE 4

within 36 hours

L01

Required Action and associated Completion Time not met or one or more steam generators with less than five MSSVs OPERABLE

SURVEILLANCE REQUIREMENTS

Add proposed SR Note

L02

SR 3.7.1.1

4.7.1.1 Verify each ~~main steam line code safety valve is OPERABLE~~ in accordance with the INSERVICE TESTING PROGRAM. Following testing, ~~as-left~~ lift settings shall be within +/- 1% ~~of 1000 psia for valves 8201 through 8208, and within +/- 1% of 1040 psia for valves 8209 through 8216 specified in Table 3.7-2~~.

LA01

required MSSV lift setpoint per Table 3.7.1-2

Table 3.7.1-1

TABLE 3.7.1

VARIABLE

VERSUS

MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT ~~WITH IN~~OPERABLE

MAIN

STEAM ~~LINE~~ SAFETY VALVES ~~DURING OPERATION WITH BOTH STEAM GENERATORS~~

Minimum MSSVs per ~~Maximum~~ Number of ~~Inoperable Safety Valves on Any Operating~~ Steam Generator	Required Operable	Variable Maximum Allowable Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
7	~~1~~	92.8
6	~~2~~	79.6
5	~~3~~	66.3

A04

3.7.1-2

TABLE ~~3.7-2~~

Main Steam Safety Valve Lift Settings

~~STEAM LINE SAFETY VALVES PER LOOP~~

Table 3.7.1-2

	VALVE NUMBER		LIFT SETTING*
	Steam Generator 2A — ~~Header A~~	Steam Generator 2B — ~~Header B~~	
Bank 1	a. 8201	8205	~~≥ 955.3 psig and ≤ 1015.3 psig~~
Bank 1	b. 8202	8206	~~≥ 955.3 psig and ≤ 1015.3 psig~~
Bank 1	c. 8203	8207	~~≥ 955.3 psig and ≤ 1015.3 psig~~
Bank 1	d. 8204	8208	~~≥ 955.3 psig and ≤ 1015.3 psig~~
Bank 2	e. 8209	8213	~~≥ 994.1 psig and ≤ 1046.1 psig~~
Bank 2	f. 8210	8214	~~≥ 994.1 psig and ≤ 1046.1 psig~~
Bank 2	g. 8211	8215	~~≥ 994.1 psig and ≤ 1046.1 psig~~
Bank 2	h. 8212	8216	~~≥ 994.1 psig and ≤ 1046.1 psig~~
	V	V	

985 psig

1025 psig

A07

* +/-3% for ~~valves a through d~~ and +2%/-3% for ~~valves e through h~~

Bank 1

Bank 2

DISCUSSION OF CHANGES
ITS 3.7.1, MAIN STEAM SAFETY VALVES (MSSVs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.7.1.1 states that “All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 4.7-1.” CTS Table 4.7-1 lists lift setting pressures for eight safety valves in each of the two loops. Unit 2 CTS 3.7.1.1 states that “All main steam line code safety valves shall be OPERABLE with lift settings as shown in Table 3.7-2.” CTS Table 3.7-2 lists lift setting pressures for eight safety valves in each of the two loops. ITS LCO 3.7.1 requires the MSSVs specified in Tables 3.7.1-1 and 3.7.1-2 to be OPERABLE. This changes the CTS by combining the current LCO requirement and portions of Unit 1 CTS Table 4.7-1 and Unit 2 CTS Table 3.7-2 into a single ITS LCO requirement for each respective unit.

This change is acceptable because the number of MSSVs required OPERABLE under the various conditions has not changed. This change results in a format change only to comply with the manner in which the ISTS presents the MSSV requirements. This change is designated as an administrative change since it does not result is any technical changes to the CTS.

A03 CTS 3.7.1.1 ACTION provides compensatory actions for one or more inoperable main steam line code safety valves. CTS 3.7.1.1 ACTION requires that within 4 hours the inoperable valve be restored to OPERABLE status or the Power Level-High trip setpoint be reduced per the requirements of CTS Table 3.7-1. CTS 3.7.1.1 ACTION requires a unit shutdown if the requirements of the ACTION are not met. ITS 3.7.1 ACTIONS Note states "Separate Condition entry is allowed for each MSSV." This changes the CTS by explicitly specifying separate condition entry for each inoperable MSSV.

The purpose of the CTS ACTION is to allow separate condition entry for each inoperable MSSV. Each time it is discovered that an MSSV is inoperable, entry is required and the specified Completion Time is allowed to complete the compensatory actions. The ITS 3.7.1 ACTIONS Note allows a separate Completion Time clock for each MSSV that is inoperable. This change is acceptable, because it only provides clarification of the Completion Time when one valve is inoperable and, subsequently, a second valve becomes inoperable. This change is designated as administrative, because it does not result in a technical change to the Specifications.

A04 CTS 3.7.1.1 ACTION states that the Power Level-High trip setpoint must be reduced per CTS Table 3.7-1 when one or more MSSVs are found to be inoperable. CTS Table 3.7-1 provides the maximum allowable Power Level-High trip setpoint corresponding to the maximum number of inoperable MSSVs on any

operating steam generator. ITS 3.7.1 ACTION A requires both a reduction in THERMAL POWER and a reduction in the Power Level-High trip setpoint consistent with the requirements of ITS Table 3.7.1-1. The Table has been revised to provide the associated maximum allowable power for the number of OPERABLE MSSVs. This changes the CTS by adding an additional explicit statement to reduce THERMAL POWER consistent with ITS Table 3.7.1-1 and by stating the maximum allowable power as a function of OPERABLE, instead of inoperable, MSSVs.

The purpose of CTS 3.7.1.1 ACTION is to reduce the Power Level-High trip setpoint to within the limits of the safety analyses. Current plant operation dictates that THERMAL POWER is reduced before reducing the setpoints to prevent a reactor trip. Explicitly stating this practice in TS and stating the maximum power level in terms of OPERABLE instead of inoperable MSSVs does not change how the plant is operated. This change is considered administrative because it does not result in technical changes to the CTS.

A05 CTS 3.7.1.1 ACTION states that with one or more MSSVs inoperable, either restore the inoperable valves to OPERABLE status or reduce the Power Level-High trip setpoint. ITS 3.7.1 ACTION A does not include the restoration requirement, only the alternate compensatory measure. This changes the CTS by eliminating the explicit statement to restore the MSSV(s) to OPERABLE status.

This change is acceptable, because it does not result in a technical change to the Technical Specifications. Restoration of compliance with the LCO is always an option in an ACTION, so eliminating the restoration ACTION from the CTS has no effect. In both the CTS and the ITS, if the inoperable MSSV(s) are not restored, actions are taken that result in reducing reactor power to within the relief capability of the OPERABLE MSSVs within 4 hours. This change is designated as administrative, because it does not result in a technical change to the CTS.

A06 CTS 3.7.1.1 ACTION requires in part, “otherwise be in… MODE 5...” when the actions cannot be performed within the required time. ITS 3.7.1 Required Action B.2 requires, when the Required Action and associated Completion Time is not met; or, with one or more steam generators with less than five MSSVs OPERABLE, to be in MODE 4 within 12 hours. This changes the end states in CTS 3.7.1.1 ACTION from MODE 5 to MODE 4.

The purpose of CTS 3.7.1 ACTION is to place the plant in a condition in which the equipment is no longer required. CTS 3.7.1 ACTION provides appropriate actions to take for the MSSVs when the unit is in MODE 1, 2, or 3. The MSSVs are not required to be OPERABLE when in MODE 4. Therefore, in accordance with CTS 3.0.2 (ITS LCO 3.0.2), with the Required Action and Completion time for one or more required MSSVs inoperable or one or more steam generators with less than five MSSVs OPERABLE completion of the Required Action to MODE 5 is not required. These changes are designated as administrative changes and are acceptable because they maintain the current requirement to place the unit in a condition in which the MSSVs are no longer required consistent with the requirements of CTS 3.0.2 (ITS LCO 3.0.2). The 12-hour Completion Time is reasonable based on operating experience to reach the required unit conditions.

A07 Unit 1 CTS Table 4.7-1 and Unit 2 CTS Table 3.7-2 specifies a lift setting band of ≥ 955.3 psig and ≤ 1015.3 psig for Bank 1 Main Steam Safety Valves and ≥ 994.1 psig and ≤ 1046.1 psig for Bank 2 Main Steam Safety Valves (MSSVs). ITS Table 3.7.1-2 a lift setting of 985 psig with a ± 3% tolerance for Bank 1 MSSVs and a lift setting of 1025 psig with a + 2%/− 3% tolerance for Bank 2 MSSVs. This changes the CTS by correcting the nominal lift setpoint to the value assumed in the overpressure protection report and documented in the UFSAR; 985 psig for Bank 1 and 1040 psig for Bank 2.

These changes are designated as administrative changes and are acceptable because they state the specific nominal lift setpoint assumed in the analysis instead of providing a band with an upper and lower value based on a nominal setpoint value.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS SR 4.7.1.1 states, in part, “as left lift settings shall be within ± 1% of 1000 psia for valves 8201 through 8208, and within ± 1% of 1040 psia for valves 8209 through 8216 specified in Table 4.7-1.” ITS 3.7.1 does not contain the detail “of 1000 psia for valves 8201 through 8208, and within ± 1% of 1040 psia for valves 8209 through 8216 specified in Table 4.7-1” for Unit 1 or “of 1000 psia for valves 8201 through 8208, and within ± 1% of 1040 psia for valves 8209 through 8216 specified in Table 3.7-2” for Unit 2. This changes the CTS by moving details on setting the lift pressure in terms of psia to the ITS Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the lift settings and the definition of OPERABLE states that the components must be capable of performing their safety function. In addition, ITS SR 3.7.1.1 continues to require lift settings to be within + 1% following testing. It is understood that the MSSVs must be adjusted to lift at the settings given under the conditions that the safety analysis assumes the MSSVs will operate. This change is acceptable, because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less

restrictive removal of detail change, because details for meeting Technical Specification requirements are being removed from the Technical Specifications to the ITS Bases.

LESS RESTRICTIVE CHANGES

L01 (*Category 3 – Relaxation of Completion Time)* CTS 3.7.1.1 ACTION specifies the compensatory actions when one or more main steam line code safety valves are inoperable. The action allows operation to continue provided that within 4 hours, either the inoperable main steam line code safety valves are restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-1. ITS 3.7.1 Required Action A.2 requires the reduction of the Power Level-High trip setpoint to less than or equal to the Maximum Allowable Power Level-High trip setpoint (Percent of RATED THERMAL POWER) specified in Table 3.7-1 within 36 hours. This changes the CTS by extending the time allowed to reduce the Power Level-High trip setpoint. The discussion of the change that deletes the restoration option is discussed in DOC A05.

The purpose of CTS 3.7.1.1 ACTION is to limit the time the unit can operate with inoperable main steam line code safety valves without reducing the Power Level-High trip setpoints. This change is acceptable, because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs, and the low probability of a DBA occurring during the allowed Completion Time. This change extends the time allowed to reduce the Power Level-High trip setpoints when the MSSVs are inoperable. The time extension is from 4 hours to 36 hours. However, the time to reduce THERMAL POWER to within the same limits is maintained in ITS 3.7.1 Required Action A.1, as described in DOC A04. This change is acceptable since the Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting the channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period. In addition, the actual reactor power level continues to be required to be reduced within the same limits within 4 hours. Thus, operation of the unit at RATED THERMAL POWER with an inoperable MSSV is still only allowed for 4 hours, consistent with the current allowance. This change is designated as less restrictive, because additional time is allowed to reduce the applicable reactor trip setpoints than was allowed in the CTS.

L02 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.7.1.1 requires verification each main steam line code safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. ITS SR 3.7.1.1 requires the same testing to verify OPERABILITY, however, a Note has been included that only requires the lift setpoint verification to be performed within its Frequency in MODES 1 and 2. This changes the CTS by adding a Note that allows the unit to enter MODE 3 without having performed the MSSV lift setpoint verification.

DISCUSSION OF CHANGES
ITS 3.7.1, MAIN STEAM SAFETY VALVES (MSSVs)

The purpose of CTS 4.7.1.1 is to perform the MSSV lift setpoint verification in accordance with the INSERVICE TESTING PROGRAM. The SR is modified by a Note that states the Surveillance is only required to be performed in MODES 1 and 2. The Note allows entry into and operation in MODE 3 prior to performing the SR. By allowing entry into MODE 3 prior to performing the SR, testing will be performed at hot conditions. This change is acceptable because the Technical Specifications continue to require the MSSVs to be OPERABLE in MODE 3. In addition, the Surveillance continues to provide an acceptable level of equipment reliability. This change is designated as less restrictive because Surveillances will be performed in fewer operating conditions than in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

Eight MSSVs per steam generator

3.7.1.1 DOC A02 LCO 3.7.1 ~~The MSSVs~~ shall be OPERABLE ~~as specified in Table 3.7.1-1 and Table 3.7.1-2~~. (3)

Applicability APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------

DOC A03 Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a DOC A04 A. One or more ~~required~~ MSSVs inoperable.	A.1 Reduce power to less than or equal to the applicable % RTP listed in Table 3.7.1-1. AND	4 hours (1)
DOC L01	Variable Power Level-High trip A.2 Reduce the ~~[variable overpower trip - high]~~ setpoint ~~[ceiling]~~ in accordance with Table 3.7.1-1.	36 hours (2)
Action a B. Required Action and associated Completion Time not met. OR DOC A06 One or more steam generators with less than ~~[two]~~ MSSVs OPERABLE. (five)	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours (2)

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.7.1.1 DOC L02	SR 3.7.1.1	----------NOTE---------- Only required to be performed in MODES 1 and 2. ----------	
		Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

CTS

Table 3.7-1

Power Level - High

Table 3.7.1-1 (page 1 of 1)
[Variable ~~Overpower~~ Trip] Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	~~MAXIMUM POWER (% RTP)~~	Variable Power Level - High MAXIMUM ALLOWABLE [VARIABLE ~~OVERPOWER~~ TRIP] SETPOINT ([~~CEILING~~] % RTP)
7 ~~[8]~~	~~[ ]~~	88.5 ~~[ ]~~
6 ~~[7]~~	~~[ ]~~	79.8 ~~[ ]~~
5 ~~[6]~~	~~[ ]~~	66.5 ~~[ ]~~
~~[5]~~	~~[ ]~~	~~[ ]~~
~~[4]~~	~~[ ]~~	~~[ ]~~
~~[3]~~	~~[ ]~~	~~[ ]~~
~~[2]~~	~~[ ]~~	~~[ ]~~

2

CTS

Table 4.7-1

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

Valve Bank	VALVE NUMBER		LIFT SETTING (~~psig ± [3]%~~)
	1A Steam Generator ~~#1~~	1B Steam Generator ~~#2~~	psig ±3% for Bank 1 psig +2%/−3% for Bank 2
1	V8201 V8202 V8203 V8204 ~~[ ]~~	V8205 V8206 V8207 V8208 ~~[ ]~~	~~[ ]~~ 985 psig ~~[ ]~~
2	V8209 V8210 V8211 V8212 ~~[ ]~~	V8213 V8214 V8215 V8216 ~~[ ]~~	~~[ ]~~ 1025 psig ~~[ ]~~

1
2
1

CTS

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

3.7.1.1 DOC A02

LCO 3.7.1 ~~The MSSVs~~ shall be OPERABLE ~~as specified in Table 3.7.1-1 and Table 3.7.1-2~~. (3)

Eight MSSVs per steam generator

Applicability

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------

DOC A03 Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a DOC A04 A. One or more ~~required~~ MSSVs inoperable.	A.1 Reduce power to less than or equal to the applicable % RTP listed in Table 3.7.1-1. AND	4 hours (1)
DOC L01	Variable Power Level-High trip A.2 Reduce the ~~[variable overpower trip - high]~~ setpoint ~~[ceiling]~~ in accordance with Table 3.7.1-1.	36 hours (2)
Action a B. Required Action and associated Completion Time not met. OR DOC A06 One or more steam generators with less than ~~[two]~~ MSSVs OPERABLE. five	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours (2)

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.7.1.1 DOC L02	SR 3.7.1.1	----------NOTE---------- Only required to be performed in MODES 1 and 2. ----------	
		Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

CTS

Table 3.7-1

Power Level - High

Table 3.7.1-1 (page 1 of 1)
[Variable ~~Overpower~~ Trip] Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	~~MAXIMUM POWER (% RTP)~~	Variable Power Level - High MAXIMUM ALLOWABLE [VARIABLE ~~OVERPOWER~~ TRIP] SETPOINT ([~~CEILING~~] % RTP)
7 ~~[8]~~	~~[ ]~~	92.8 ~~[ ]~~
6 ~~[7]~~	~~[ ]~~	79.6 ~~[ ]~~
5 ~~[6]~~	~~[ ]~~	66.3 ~~[ ]~~
~~[5]~~	~~[ ]~~	~~[ ]~~
~~[4]~~	~~[ ]~~	~~[ ]~~
~~[3]~~	~~[ ]~~	~~[ ]~~
~~[2]~~	~~[ ]~~	~~[ ]~~

2

CTS

Table 4.7-1

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

Valve Bank	VALVE NUMBER		LIFT SETTING (~~psig ± [3]%~~)
	2A Steam Generator ~~#1~~	2B Steam Generator ~~#2~~	psig ±3% for Bank 1 psig +2%/−3% for Bank 2
1	V8201 V8202 V8203 V8204	V8205 V8206 V8207 V8208	~~[ ]~~ 985 psig ~~[ ]~~
2	V8209 V8210 V8211 V8212	V8213 V8214 V8215 V8216	~~[ ]~~ 1025 psig ~~[ ]~~

1
2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.1, MAIN STEAM SAFETY VALVES (MSSVs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. ISTS LCO 3.7.1 refers to Tables 3.7.1-1 and 3.7.1-2 for MSSV OPERABILITY. Table 3.7.1-1 does not provide information related to MSSV OPERABILITY but rather provides the allowable power level for the THERMAL POWER and trip setpoint reductions as referenced in Required Actions A.1 and A.2. In addition, SR 3.7.1.1 references Table 3.7.1-2 for the MSSV lift settings. Therefore, reference to the tables in the LCO is unnecessary and is changed in the ITS as a presentation preference. ITS LCO 3.7.1 states that eight MSSVs per steam generator are required to be OPERABLE. This presentation preference is also consistent with the presentation used in LCO 3.7.1 of the Westinghouse Engineering ISTS (NUREG-1431, Revision 5).

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing ~~a heat sink~~ for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available. 1

Eight MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section ~~[~~5.2~~]~~ (Ref. 1). The MSSV ~~rated capacity passes the full steam flow at 102% RTP (100% + 2% for instrument error) with the valves full open~~. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-~~1~~, in the accompanying LCO, so that only the number of valves needed will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open all valves following a turbine reactor trip. 1

must have sufficient capacity to limit the secondary system pressure to ≤ 110% of design pressure

U

s

2

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2. The MSSV's purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing ~~100~~% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis. 1

100.3

The events that challenge the MSSV relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in the FSAR, Section ~~[~~15.2~~]~~ (Ref. 3). Of these, the ~~full power loss of condenser vacuum (LOCV) event is the limiting AOO. An LOCV isolates the turbine and condenser, and terminates normal feedwater flow to the steam generators. Before delivery of auxiliary feedwater to the steam generators,~~ RCS pressure reaches ≤ ~~2630 psig~~. This peak pressure is < 110% of the design pressure of 2500 ~~psig~~, but high enough to actuate the pressurizer safety valves. The maximum relieving rate ~~during the LOCV event is 2.5 E6 lb/hour, which is less than~~ the rated capacity of ~~two~~ MSSVs. 2 1

U

limiting and bounding event is a loss of external load (LOEL),

2744 psia

psia

is within

the

The LOEL event analysis demonstrates that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure. The peak main steam system pressure (steam generator dome) and peak secondary side pressure for the limiting case are less than 110% of design. The peak main steam system pressure (steam generator dome) is 1092 psia.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The first bank of four MSSVs per steam generator are assumed to lift at 1000 psia (985 psig) with a maximum tolerance of + 3% and the second bank of four MSSVs per steam generator lift at 1040 psia (1025 psig) with a maximum tolerance of + 2%.

~~The limiting accident for peak RCS pressure is the full power feedwater line break (FWLB), inside containment, with the failure of the backflow check valve in the feedwater line from the affected steam generator. Water from the affected steam generator is assumed to be lost through the break with minimal additional heat transfer from the RCS. With heat removal limited to the unaffected steam generator, the reduced heat transfer causes an increase in RCS temperature, and the resulting RCS fluid expansion causes an increase in pressure. The RCS pressure increases to ≤ 2730 psig, with the pressurizer safety valves providing relief capacity. The maximum relieving rate of the MSSVs during the FWLB event is ≤ 2.5 E6 lb/hour, which is less than the rated capacity of two MSSVs.~~

~~Using conservative analysis assumptions, a small range of FWLB sizes less than a full double ended guillotine break produce an RCS pressure of 2765 psig for a period of 20 seconds; exceeding 110% (2750 psig) of design pressure. This is considered acceptable as RCS pressure is still well below 120% of design pressure where deformation may occur. The probability of this event is in the range of 4 E-6/year.~~

1

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

. The LCO is met when eight MSSVs per steam generator are OPERABLE. Operation with less than the required number of MSSVs per steam generator OPERABLE

This LCO requires all MSSVs to be OPERABLE ~~in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with less than the full number of MSSVs~~ requires limitations on allowable THERMAL POWER (to meet Reference 2 requirements), and adjustment to the Reactor Protection ~~n~~ ve System trip setpoints. These limitations are according to those shown in Table 3.7.1-1, Required Action A.1, and Required Action A.2 in the ~~accompanying LCO~~ ACTIONS. An MSSV is considered inoperable if it fails to open upon demand.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the INSERVICE TESTING PROGRAM.

The lift settings, according to Table 3.7.1-2 ~~in the accompanying LCO,~~ correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the RCPB.

1

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BASES

s 1, 2, and 3

five

APPLICABILITY

In MODE ~~1,~~ a minimum of ~~two~~ MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 ~~in the accompanying LCO,~~ which is limiting and bounds all lower MODES. ~~In MODES 2 and 3, both the ASME Code and the accident analysis require only one MSSV per steam generator to provide overpressure protection.~~

1

In MODES 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the steam generators is no greater than the available relief capacity in that steam generator.

Operation at or below the maximum power will ensure the design overpressure limits will not be exceeded.

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. Operation may continue provided ~~the allowable THERMAL POWER is equal to the product of: 1) the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator, and 2) the ratio of the available relieving capacity to total steam flow, multiplied by 100%.~~

~~Allowable THERMAL POWER = (8 - N) x 109.2~~
~~8~~

Power Level-High trip setpoint

. The reduced reactor trip allowable values are based on a detailed analysis of the LOEL with a concurrent single failure event. (Ref. 3).

With one or more MSSVs inoperable, the ~~ceiling on the variable overpower trip~~ is reduced ~~to an amount over the allowable THERMAL POWER equal to the band given for this trip, according to Table 3.7.1-1 in the accompanying LCO.~~

1

~~SP = Allowable THERMAL POWER + 9.8~~

~~where:~~

~~SP = Reduced reactor trip setpoint in percent RTP. This is a ratio of the available relieving capacity over the total steam flow at rated power.~~

~~8 = Total number of MSSVs per steam generator.~~

~~N = Number of inoperable MSSVs on the steam generator with the greatest number of inoperable valves.~~

1

BASES

ACTIONS (continued)

~~109.2 = Ratio of MSSV relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% RTP + 2% instrument uncertainty expressed as a percentage (see text above).~~

~~9.8 = Band between the maximum THERMAL POWER and the variable overpower trip setpoint ceiling (Table 3.7.1-1).~~

1

The operator should limit the maximum steady state power level to some value slightly below this setpoint to avoid an inadvertent overpower trip.

The 4 hour Completion Time for Required Action A.1 is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 32 hours is allowed in Required Action A.2 to reduce the setpoints. The Completion Time of 36 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time, or if one or more steam generators have less than ~~two~~ five MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within ~~[~~12~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the INSERVICE TESTING PROGRAM. The ASME Code (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM~~-1-1987~~ Code 2004/2006a (Ref. 5). According to Reference 5, the following tests are required for MSSVs:

1

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

a. Visual examination,

b. Seat tightness determination,

c. Setpoint pressure determination (lift setting), and

d. Compliance with owner's seat tightness criteria~~, and~~

~~e. Verification of the balancing device integrity on balanced valves~~.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± ~~[~~3~~]~~% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

for the lower pressure setting valves and +2%/−3% setpoint tolerance for the higher pressure setting valves

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This is to allow testing of the MSSVs at hot conditions. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section ~~[~~5.2~~]~~. (U)
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. FSAR, Section ~~[~~15.2~~]~~. (U)
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. ANSI/ASME OM~~ 1-1987~~. (Code 2004/2006a)

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing ~~a heat sink~~ for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available. 1

Eight MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section ~~[~~5.2~~]~~ (Ref. 1). The MSSV ~~rated capacity passes the full steam flow at 102% RTP (100% + 2% for instrument error) with the valves full open~~. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-~~1~~, in the accompanying LCO, so that only the number of valves needed will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open all valves following a turbine reactor trip. 1

must have sufficient capacity to limit the secondary system pressure to ≤ 110% of design pressure

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APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2. The MSSV's purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing ~~100~~% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis. 1

100.3

The events that challenge the MSSV relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in the FSAR, Section ~~[~~15.2~~]~~ (Ref. 3). Of these, the ~~full power loss of condenser vacuum (LOCV) event is the limiting AOO. An LOCV isolates the turbine and condenser, and terminates normal feedwater flow to the steam generators. Before delivery of auxiliary feedwater to the steam generators,~~ RCS pressure reaches ≤ ~~2630 psig~~. This peak pressure is < 110% of the design pressure of 2500 ~~psig~~, but high enough to actuate the pressurizer safety valves. The maximum relieving rate ~~during the LOCV event is 2.5 E6 lb/hour, which is less than~~ the rated capacity of ~~two~~ MSSVs. 2 1

limiting and bounding event is a Loss of Condenser Vacuum (LOCV),

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2669 psia

psia

is within

the

The LOCV event analysis demonstrates that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure. The peak main steam system pressure (steam generator dome) and peak secondary side pressure for the limiting case are less than 110% of design. The peak main steam system pressure (steam generator dome) is 1094.6 psia. | R1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The first bank of four MSSVs per steam generator are assumed to lift at 1000 psia (985 psig) with a maximum tolerance of + 3% and the second bank of four MSSVs per steam generator lift at 1040 psia (1025 psig) with a maximum tolerance of + 2%.

~~The limiting accident for peak RCS pressure is the full power feedwater line break (FWLB), inside containment, with the failure of the backflow check valve in the feedwater line from the affected steam generator. Water from the affected steam generator is assumed to be lost through the break with minimal additional heat transfer from the RCS. With heat removal limited to the unaffected steam generator, the reduced heat transfer causes an increase in RCS temperature, and the resulting RCS fluid expansion causes an increase in pressure. The RCS pressure increases to ≤ 2730 psig, with the pressurizer safety valves providing relief capacity. The maximum relieving rate of the MSSVs during the FWLB event is ≤ 2.5 E6 lb/hour, which is less than the rated capacity of two MSSVs.~~

1

~~Using conservative analysis assumptions, a small range of FWLB sizes less than a full double ended guillotine break produce an RCS pressure of 2765 psig for a period of 20 seconds; exceeding 110% (2750 psig) of design pressure. This is considered acceptable as RCS pressure is still well below 120% of design pressure where deformation may occur. The probability of this event is in the range of 4 E-6/year.~~

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

. The LCO is met when eight MSSVs per steam generator are OPERABLE. Operation with less than the required number of MSSVs per steam generator OPERABLE

This LCO requires all MSSVs to be OPERABLE ~~in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with less than the full number of MSSVs~~ requires limitations on allowable THERMAL POWER (to meet Reference 2 requirements), and adjustment to the Reactor Protection ~~n~~ ve System trip setpoints. These limitations are according to those shown in Table 3.7.1-1, Required Action A.1, and Required Action A.2 in the ~~accompanying LCO~~. An MSSV is considered inoperable if it fails to open upon demand. ACTIONS

1

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the INSERVICE TESTING PROGRAM.

The lift settings, according to Table 3.7.1-2 ~~in the accompanying LCO,~~ correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the RCPB.

BASES

APPLICABILITY

In MODE ~~1,~~ a minimum of ~~two~~ MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODES. ~~In MODES 2 and 3, both the ASME Code and the accident analysis require only one MSSV per steam generator to provide overpressure protection.~~

In MODES 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the Steam Generators is no greater than the available relief capacity in that Steam Generator.

Operation at or below the maximum power will ensure the design overpressure limits will not be exceeded.

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. Operation may continue provided ~~the allowable THERMAL POWER is equal to the product of: 1) the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator, and 2) the ratio of the available relieving capacity to total steam flow, multiplied by 100%.~~

~~$$\text{Allowable THERMAL POWER} = \frac{(8 - N)}{8} \times 109.2$$~~

. The reduced reactor trip allowable values are based on a detailed analysis of the LOCV event with a concurrent single failure event. (Ref. 3).

With one or more MSSVs inoperable, the ~~ceiling on the variable overpower trip~~ is reduced ~~to an amount over the allowable THERMAL POWER equal to the band given for this trip, according to Table 3.7.1-1 in the accompanying LCO.~~

~~SP = Allowable THERMAL POWER + 9.8~~

~~where:~~

~~SP = Reduced reactor trip setpoint in percent RTP. This is a ratio of the available relieving capacity over the total steam flow at rated power.~~

~~8 = Total number of MSSVs per steam generator.~~

~~N = Number of inoperable MSSVs on the steam generator with the greatest number of inoperable valves.~~

BASES

ACTIONS (continued)

~~109.2 = Ratio of MSSV relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% RTP + 2% instrument uncertainty expressed as a percentage (see text above).~~

~~9.8 = Band between the maximum THERMAL POWER and the variable overpower trip setpoint ceiling (Table 3.7.1-1).~~

1

The operator should limit the maximum steady state power level to some value slightly below this setpoint to avoid an inadvertent overpower trip.

The 4 hour Completion Time for Required Action A.1 is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 32 hours is allowed in Required Action A.2 to reduce the setpoints. The Completion Time of 36 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time, or if one or more steam generators have less than ~~two~~ five MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within ~~[~~12~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

1

2

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the INSERVICE TESTING PROGRAM. The ASME Code (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM~~-1-1987~~ Code 2004/2006a (Ref. 5). According to Reference 5, the following tests are required for MSSVs:

1

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

a. Visual examination,

b. Seat tightness determination,

c. Setpoint pressure determination (lift setting), and

d. Compliance with owner's seat tightness criteria~~, and~~

~~e. Verification of the balancing device integrity on balanced valves~~.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± ~~[~~3~~]~~% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

for the lower pressure setting valves and +2%/−3% setpoint tolerance for the higher pressure setting valves

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This is to allow testing of the MSSVs at hot conditions. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

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1. FSAR, Section ~~[~~5.2~~]~~.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

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3. FSAR, Section ~~[~~15.2~~]~~.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. ANSI/ASME OM~~ 1-1987~~. Code 2004/2006a

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.1 BASES, MAIN STEAM SAFETY VALVES (MSSVs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is changed to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.1, MAIN STEAM SAFETY VALVES (MSSVs)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

ITS 3.7.2, Main Steam Isolation Valves (MSIVs)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

PLANT SYSTEMS

MAIN STEAM ~~LINE~~ ISOLATION VALVES

(MSIVs)

LIMITING CONDITION FOR OPERATION

LCO 3.7.2 3.7.1.5 Two MSIVs ~~Each main steam line isolation valve~~ shall be OPERABLE.

Applicability **APPLICABILITY:** MODES 1, ~~2 and 3.~~ MODES 2 and 3 except when all MSIVs are closed. L01

ACTION:

Action A **MODE 1** - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is ~~either~~ restored to OPERABLE status ~~or closed~~ within 8 ~~4~~ hours; otherwise, be in MODE 2 within ~~the next~~ 6 hours. M01 L04

Action B **MODES 2 and 3** - With one or ~~both~~ main steam isolation valve(s) inoperable, subsequent operation in MODES 2 or 3 may continue provided: Add proposed Condition C Note more A02

Action C
1. The inoperable main steam isolation valves are closed within 8 hours, and
2. The inoperable main steam isolation valves are verified closed once per 7 days.

Action D Otherwise, be in at least ~~HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 24~~ 12 hours. MODE 3 MODE 4 M02

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.7.2.1 Note L02

SR 3.7.2.1 4.7.1.5 ~~Each main steam line isolation valve that is open shall be demonstrated OPERABLE by~~ V ~~verifying full closure~~ within ~~6.0 seconds when tested pursuant to~~ the INSERVICE TESTING PROGRAM. limits in accordance with the isolation time of each MSIV is LA01

Add proposed SR 3.7.2.2 and Note M03

~~Pages 3/4 7-11 through 3/4 7-12 (Amendment No. 86) have been deleted from the Technical Specifications. The next page is 3/4 7-13.~~

PLANT SYSTEMS

MAIN STEAM ~~LINE~~ ISOLATION VALVES

(MSIVs)

LIMITING CONDITION FOR OPERATION

LCO 3.7.2 3.7.1.5 Two MSIVs ~~Each main steam line isolation valve~~ shall be OPERABLE.

Applicability **APPLICABILITY:** MODES 1, ~~2, 3 and 4.~~ L03 L01

MODES 2 and 3 except when all MSIVs are closed.

ACTION:

Action A **MODE 1** – With one main steam line isolation valve inoperable ~~but open~~, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 8 hours; otherwise, be in at least MODE 2 within ~~the next~~ 6 hours. M01 L04

Action B

Add proposed Condition C Note

A02

Action C **MODES 2, 3 and 4** – With one or ~~both~~ main steam isolation valve(s) inoperable, subsequent operation in MODES 2~~,~~ 3 ~~or 4~~ may proceed provided: L03

more

1. The inoperable main steam isolation valves are closed within 8 hours, and

2. The inoperable main steam isolation valves are verified closed once per 7 days.

Action D Otherwise, be in at least ~~HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following 24~~ hours.

MODE 3

MODE 4

12

L03

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.7.2.1 Note

V

L02

SR 3.7.2.1 4.7.1.5 ~~Each main steam line isolation valve shall be demonstrated OPERABLE by v~~erify~~ing full closure~~ within ~~6.75 seconds when tested pursu~~ant to the INSERVICE TESTING PROGRAM. LA01

limits in accordance with

the isolation time of each MSIV is

M03

Add proposed SR 3.7.2.2 and Note

DISCUSSION OF CHANGES
ITS 3.7.2, MAIN STEAM ISOLATION VALVES (MSIVs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.7.1.5 Action for MODES 2 and 3 and Unit 2 CTS 3.7.1.5 Action for MODES 2, 3 and 4 requires entry when one or both main steam isolation valves are inoperable. ITS 3.7.2 ACTION C includes a Condition Note that specifies separate Condition entry is allowed for each MSIV. This changes the CTS by explicitly specifying separate entry Condition for each inoperable MSIV.

The purpose of the Unit 1 CTS 3.7.1.5 Action for MODES 2 and 3 and Unit 2 CTS 3.7.1.5 Action for MODES 2, 3 and 4 is to ensure the appropriate compensatory actions are in place for when one or more MSIVs are not OPERABLE. This change is acceptable because the intent of the CTS Action is to allow separate entry for each inoperable MSIV. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.7.1.5 Action for MODE 1 compensatory measure states, in part, “provided the inoperable valve is either restored to OPERABLE status or closed” and Unit 2 CTS 3.7.1.5 Action for MODE 1 compensatory measure states when one MSIV is inoperable "but open." ITS 3.7.2 ACTION A provides compensatory actions for when an MSIV is inoperable, regardless of whether the valve is open or closed. This changes Unit 1 CTS 3.7.1.5 by deleting the compensatory measure allowing closure of the inoperable valve to fully satisfy the Action and changes the Unit 2 CTS 3.7.1.5 condition for entry into the action from "inoperable but open" to "inoperable."

The purpose of the CTS 3.7.1.5 Action for MODE 1 is to ensure that appropriate compensatory actions are in place when an MSIV valve will not close within the time specified. This change is necessary because the proposed Condition requires entry regardless of whether the MSIV is open or closed. Operation in MODE 1 with a steam generator isolated would result in an asymmetrical steam generator reactor trip. In addition, two Reactor Coolant System (RCS) loops are required to be in operation in MODE 1 and if an MSIV is closed, the steam generator would not be performing its design function to provide a heat sink to the associated RCS loop. The closure of the MSIV could cause the associated main steam safety valves to open with an associated transient requiring operator action. Nevertheless, if an MSIV is found inoperable, entry into the Condition would still be necessary in MODE 1 because MODE 1 operation cannot continue with a closed MSIV. ITS 3.7.2 ACTION A requires restoring the MSIV to

OPERABLE status within 8 hours. This change is acceptable because the MSIVs are associated with a containment penetration with a closed system (i.e., steam generator tubes). The Completion Time is consistent with safe operation under the specified Condition, the capacity and capability of remaining features (i.e., a closed system), a reasonable time for repairs of required features, and the low probability of a DBA occurring during the allowed Completion Time. This change is designated as more restrictive since it requires entry into the Condition regardless of the status (open or closed) of the inoperable MSIV and does not allow closure of the MSIV as a remedial action in MODE 1. (See DOC L04 for discussion Completion Time change.)

M02 **Unit 1 only:** CTS 3.7.1.5 Action requires, when other actions cannot be performed, to be in HOT STANDBY (MODE 3) within 6 hours and in HOT SHUTDOWN (MODE 4) within the following 24 hours. When a shutdown of the unit is required due to one or more inoperable MSIVs, ITS 3.7.2 ACTION D requires the unit to be in MODE 3 within 6 hours and MODE 4 within 12 hours. This changes the CTS Action from MODE 4 within the following 24 hours (i.e., 30 hours total) to within 12 hours.

The purpose of the time to reach the lower MODES of operation is to reach from full power conditions in an orderly manner and without challenging plant systems. The proposed Completion Time to reach MODE 4 is consistent with the other CTS actions to reach Hot Shutdown (e.g., CTS 3.6.1.3, Action a.3) and consistent with the ISTS. This change is designated as more restrictive because it requires less time to reach Hot Shutdown conditions than in the CTS.

M03 The CTS does not require testing to verify that the MSIVs close on an actuation signal. ITS SR 3.7.2.2 requires verification that each MSIV actuates to the isolation position on an actual or simulated actuation signal with a Frequency in accordance with the Surveillance Frequency Control Program (SFCP). A Note is included that requires the Surveillance to only be performed in MODES 1 and 2. This changes the CTS by requiring verification that each MSIV actuates to the isolation position on an actual or simulated actuation signal.

The purpose of the ITS SR 3.7.2.2 is to verify the MSIV can close on an actual or simulated actuation signal. This change is acceptable because the test is periodically conducted to ensure that the MSIV will perform its safety function. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, SR 3.7.2.2 will be performed at a periodic Frequency in accordance with the SFCP with an initial Frequency of 18 months consistent with ISTS SR 3.7.2.2. This Frequency based on the refueling cycle. Industry operating experience has shown that these components usually pass the Surveillance when performed at an 18 month Frequency. The SFCP was established as described in FPL (PSL Unit 1 and Unit 2) “Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program” (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the

St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066).

The Note to ITS SR 3.7.2.2 allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, in order to establish conditions consistent with those under which the acceptance criterion was generated. As specified in ITS Section 1.4, once the unit reaches MODE 1 or 2, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed.

This change is designated as more restrictive because it adds additional requirements to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* Unit 1 CTS 4.7.1.5 requires verification that the closure time of each MSIV is ≤ 6.0 seconds. Unit 2 CTS 4.7.1.5 requires verification that the closure time of each MSIV is ≤ 6.75 seconds. ITS SR 3.7.2.1 requires verification that the closure time of each MSIV is within limits. This changes the CTS by moving the MSIV closure time limit to the UFSAR. | R1 | R1

The removal of MSIV closure times from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications in order to provide adequate protection of public health and safety. The ITS retains the requirement to verify that the isolation time of each MSIV is within limits. Also, this change is acceptable because these types of details will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a detail for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* Unit 1 CTS 3.7.1.5 is applicable in MODES 1, 2, and 3. Unit 2 CTS 3.7.1.5 is applicable, in part, in MODES 1, 2, and 3. ITS LCO 3.7.2 is applicable in MODE 1, and in MODES 2 and 3 except when all MSIVs are closed. This changes the CTS by making the Specification not applicable in MODES 2 and 3 when all MSIVs are closed.

The purpose of the CTS 3.7.1.5 Applicability is to ensure the MSIVs can be isolated in the event of a high energy secondary system pipe rupture (e.g., main steam line break (MSLB)). The exception to MODE 2 and 3 is added to clarify that the MSIVs are not required to be OPERABLE when they are in a position

that supports the safety analyses. When the valves are in the closed position, they are in their assumed accident position. This change is consistent with the ISTS. This change is acceptable, because when the MSIVs are closed, the adverse effects of a high energy secondary system pipe rupture are precluded and the requirements continue to ensure that the structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses. This change is designated as less restrictive because the ITS LCO requirements are applicable in fewer operating conditions than in the CTS.

L02 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.7.1.5 requires the full closure time of each main steam isolation valve verified to be within the specified isolation time when tested pursuant to the INSERVICE TESTING PROGRAM. ITS SR 3.7.2.1 requires a similar test. However, a Note modifies the SR, stating the SR is only required to be performed in MODES 1 and 2. This changes the CTS by requiring Surveillance Requirement performance only in MODES 1 and 2; allowing entry into MODE 3 without performing the Surveillance Requirement.

The purpose of CTS 4.7.1.5 is to demonstrate that the closure time of each MSIV is within the limits assumed in the containment and accident analyses. This test should normally be conducted in MODE 3 with the unit at operating temperature and pressure. Addition of the Note modifying the Surveillance Requirement allows a delay in testing until plant temperature and pressure conditions are established consistent with the conditions assumed for the acceptance criterion basis. As specified in ITS Section 1.4, once the unit reaches MODE 1 or 2, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. The CTS change is designated as less restrictive because the ITS Surveillance Requirement is required to be performed in fewer operating conditions than in the CTS.

L03 **Unit 2 only:** *(Category 2 – Relaxation of Applicability)* CTS 3.7.1.5 is applicable, in part, in MODE 4. ITS LCO 3.7.2 is applicable in MODE 1, and in MODES 2 and 3 except when all MSIVs are closed. As a result of the elimination of MODE 4 from the Applicability, ITS 3.7.2 ACTION D requires, in part, for the unit to be in MODE 4 within 12 hours. This changes the CTS by deleting the requirement for the Specification to be applicable in MODE 4. As a result, this also changes the action end state from Cold Shutdown (MODE 5) to MODE 4 with a Completion Time of 12 hours.

The purpose of the CTS 3.7.1.5 Applicability is to ensure the MSIVs can be isolated in the event of a high energy secondary system pipe rupture (e.g., main steam line break (MSLB)). In MODE 4, the steam generator and main steam line energy is low and, therefore, MSIV isolation is not required. This change is consistent with the ISTS. This change is acceptable, because the MSIV requirements continue to ensure that the structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses. In addition, the deletion of the MODE 4 Applicability is acceptable due to the low steam generator and main steam line energy and the low probability of high energy secondary system pipe rupture occurring in MODE 4.

The change in the end state is acceptable because once the unit reaches MODE 4, the MSIVs are no longer required to be OPERABLE. Therefore, it is unnecessary to cool the unit to MODE 5 conditions. The proposed time to reach MODE 4 is consistent with other CTS Actions to reach MODE 4 (e.g., CTS 3.7.1.6 Actions a and b) and the ISTS. The proposed Completion Time is reasonable, based on operating experience, to reach MODE 4 from full power conditions in an orderly manner and without challenging plant systems. This change is designated as less restrictive because the ITS LCO requirements are applicable in fewer operating conditions than in the CTS and the Action end state is changed from Cold Shutdown (MODE 5) to MODE 4.

L04 *(Category 3 – Relaxation of Completion Time)* CTS 3.7.1.5 Action MODE 1 requires restoration of an inoperable MSIV to OPERABLE status or closure of the MSIV within 4 hours. ITS 3.7.2 ACTION A requires restoring the MSIV to OPERABLE status within 8 hours. (See DOC M01 for further discussion.) This increases the CTS action time to restore an inoperable MSIV from 4 hours to 8 hours consistent with the ISTS.

This change is acceptable because the MSIVs are associated with a containment penetration with a closed system (i.e., steam generator tubes). The Completion Time is consistent with safe operation under the specified Condition, the capacity and capability of remaining features (i.e., a closed system), a reasonable time for repairs of required features, and the low probability of a DBA occurring during the allowed Completion Time. The Completion Time of 8 hours is reasonable considering the relative stability of the closed system to act as a penetration isolation boundary. This change is designated as less restrictive because the time allowed for restoration of an MSIV to OPERABLE status has been increased.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

3.7.1.5 LCO 3.7.2 ~~[~~Two~~]~~ MSIVs shall be OPERABLE. (2)

Applicability DOC L01 APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed ~~[and de-activated]~~. (2)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action MODE 1 DOC M01 DOC L04 A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	~~[~~8~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ (2)
Action MODE 1 B. Required Action and Associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
Action MODEs 2 and 3 DOC A02 C. ----------NOTE---------- Separate Condition entry is allowed for each MSIV. ---------- One or more MSIVs inoperable in MODE 2 or 3.	C.1 Close MSIV. AND C.2 Verify MSIV is closed.	~~[~~8~~]~~ hours (2) Once per 7 days
Action MODEs 2 and 3 DOC M02 D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours (2)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.7.1.5 DOC L02	SR 3.7.2.1	----------NOTE---------- Only required to be performed in MODES 1 and 2. ---------- Verify the isolation time of each MSIV is within limits.	In accordance with the INSERVICE TESTING PROGRAM
DOC M03	SR 3.7.2.2	----------NOTE---------- Only required to be performed in MODES 1 and 2. ---------- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

3.7.1.5 LCO 3.7.2 ~~[~~Two~~]~~ MSIVs shall be OPERABLE. 2

Applicability DOC L01 DOC L03 APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed ~~[and de-activated]~~. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action MODE 1 DOC M01 DOC L04 A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	~~[~~8~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ 2
Action MODE 1 B. Required Action and Associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
Action MODEs 2 and 3 DOC A02 DOC L02 C. ----------NOTE---------- Separate Condition entry is allowed for each MSIV. ---------- One or more MSIVs inoperable in MODE 2 or 3.	C.1 Close MSIV. AND C.2 Verify MSIV is closed.	~~[~~8~~]~~ hours 2 Once per 7 days
Action MODEs 2 and 3 DOC L03 D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours ~~[~~12~~]~~ hours 2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.7.1.5 DOC L02	SR 3.7.2.1	----------NOTE---------- Only required to be performed in MODES 1 and 2. ---------- Verify the isolation time of each MSIV is within limits.	In accordance with the INSERVICE TESTING PROGRAM
DOC M03	SR 3.7.2.2	----------NOTE---------- Only required to be performed in MODES 1 and 2. ---------- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.2, MAIN STEAM ISOLATION VALVES (MSIVs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs), atmospheric dump valves, and auxiliary feedwater pump turbine steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

in either steam generator

(MSIS)

The MSIVs close on a main steam isolation signal generated by ~~either~~ low steam generator pressure ~~or high containment pressure~~. ~~The MSIVs fail closed on loss of control or actuation power.~~ The MSIS also actuates the main feedwater isolation valves (MFIVs) to close. The MSIVs may also be actuated manually.

During normal at power operation the MSIVs fail open on loss of power to the solenoid actuator due to residual hydraulic pressure. On a loss of instrument air pressure, the MSIV remains open for at least 8 hours on residual hydraulic pressure.

1

U

A description of the MSIVs is found in the FSAR, Section ~~[~~10.3~~]~~ (Ref. 1).

1

2

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, as discussed in the FSAR, Section ~~[~~6.2~~]~~ (Ref. 2). It is also influenced by the accident analysis of the SLB events presented in the FSAR, Section ~~[15.1.5]~~ (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure ~~(e.g., the failure of one MSIV to close on demand)~~.

U

U

15.4.6

1

2

full

The limiting case for the containment analysis is the hot ~~zero~~ power SLB inside containment with a ~~loss of offsite power following turbine trip, and~~ failure of ~~the MSIV on the affected steam generator to close~~. ~~At zero power, the steam generator inventory and temperature are at their maximum, maximizing~~ the analyzed mass and energy release to the containment. ~~Due to reverse flow, failure of the MSIV to close contributes to the total release of the additional mass and energy in the steam headers, which are downstream of the other MSIV.~~ With the most reactive rod ~~cluster control assembly~~ assumed stuck in the fully

one containment spray pump.

Full power conditions maximize

MSIVs are assumed in the analysis to function as designed.

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the borated water injection delivered by the Emergency Core Cooling System. Other failures considered are the failure of an MFIV to close, and failure of a ~~emergency diesel generator to start~~.

main feedwater pump (MFP) to trip

The cases considered for in-containment MSLB analysis include various power levels ranging from 0 to 100.3% power. Cases for SLBs inside and outside containment were analyzed with offsite and control room dose within the regulatory acceptance criteria

~~The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip.~~

The analysis assumes

~~With offsite power available,~~ the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the ~~high pressure safety injection (HPSI)~~ pumps, is delayed. Significant single failures considered include: failure of a ~~MSIV~~ to close, failure of a~~n emergency diesel generator~~, and failure of a ~~HPSI~~ pump.

containment spray

MFIV

containment spray

MFP to trip

to close

1

3

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

The location of this break is at the steam generator outlet nozzle, upstream of the MSIVs. This location results in the largest possible steam flow for a given break size. The blowdown is limited to one steam generator due to the reverse flow check valve, which prevents flow from the intact steam generator. A MSIS occurs on low steam generator pressure to initiate the closure of the MSIVs.

a. An HELB inside containment. ~~In order to maximize the mass and energy release into the containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator, and from the residual steam in the main steam header downstream of the closed MSIV in the intact loop~~.

b. A break outside of containment and upstream from the MSIVs. This scenario is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break, and limits the blowdown to a single steam generator.

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. A break downstream of the MSIVs. This type of break will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the steam bypass valves will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIV~~[~~s~~]~~ isolates the affected steam generator from the intact steam generator. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture. (2)

e. The MSIVs are also utilized during other events such as a feedwater line break. These events are less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the MSIV in each of the ~~[~~two~~]~~ steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal. (2)

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the ~~10 CFR 100~~ (Ref. 4) limits ~~or the NRC staff approved licensing basis~~. (1) 10 CFR 50.67

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1 and in MODES 2 and 3 except when all MSIVs are closed ~~and [deactivated]~~. In these MODES there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing their safety function. (1) (2)

~~In MODE 4, the steam generator energy is low; therefore, the MSIVs are not required to be OPERABLE.~~

4, is low

In MODES 5 and 6, the steam generator~~s do not contain much~~ energy ~~because their temperature is below the boiling point of water~~; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES. (1)

BASES

ACTIONS

A.1

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the unit hot. The ~~[~~8~~]~~ hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 2

The ~~[~~8~~]~~ hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation. 2

B.1

If the MSIV cannot be restored to OPERABLE status within ~~[~~8~~]~~ hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2, and close the MSIVs in an orderly manner and without challenging unit systems. 2

C.1~~, C.2.1,~~ and C.2~~.2~~ 4

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The ~~[~~8~~]~~ hour Completion Time is consistent with that allowed in Condition A. 2

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

BASES

ACTIONS (continued)

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within ~~[~~12~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is within the limit given in Reference ~~5~~ and is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the INSERVICE TESTING PROGRAM. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. ~~6~~), requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the INSERVICE TESTING PROGRAM.

This test is conducted in MODE 3, with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, in order to establish conditions consistent with those under which the acceptance criterion was generated.

This test is conducted in MODE 3, with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, in order to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. ~~[The Frequency of MSIV testing is every [18] months. The [18] month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

REFERENCES

1. FSAR, Section ~~[~~10.3~~]~~.

2. FSAR, Section ~~[~~6.2~~]~~.

3. FSAR, Section ~~[15.1. 5]~~.

4. ~~10 CFR 100.11~~.

~~5. [Technical Requirements Manual].~~

~~6~~. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs), atmospheric dump valves, and auxiliary feedwater pump turbine steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal (MSIS) generated by either low steam generator pressure or high containment pressure. The MSIVs fail ~~closed~~ on loss of control ~~or actuation~~ power. The MSIS also actuates the main feedwater isolation valves (MFIVs) to close. The MSIVs may also be actuated manually.

MSIVs fail closed on a loss of Instrument Air.

A description of the MSIVs is found in the FSAR, Section ~~[~~10.3~~]~~ (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, as discussed in the FSAR, Section ~~[~~6.2~~]~~ (Ref. 2). It is also influenced by the accident analysis of the SLB events presented in the FSAR, Section ~~[15.1.5]~~ 15.1.6 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

. The failure of one train of the containment heat removal system with offsite power available is the most limiting active failure.

The limiting case for the containment analysis is the hot zero power SLB inside containment ~~with a loss of offsite power following turbine trip, and failure of the MSIV on the affected steam generator to close.~~ At zero power, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. ~~Due to reverse flow, failure of the MSIV to close contributes to the total release of the additional mass and energy in the steam headers, which are downstream of the other MSIV.~~ With the most reactive rod cluster control assembly assumed stuck in the fully

MSIVs are assumed in the analysis to function as designed. Initial conditions assume all rods are in the fully withdrawn position with the

BASES

APPLICABLE SAFETY ANALYSES (continued)

withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the borated water injection delivered by the Emergency Core Cooling System. Other failures considered are the failure of an MFIV to close, and failure of a emergency diesel generator to start.

or safety injection tank actuation

1

The cases considered for in-containment MSLB analysis include various power levels ranging from 0 to 100.3% power. Cases for SLBs inside and outside containment were analyzed with offsite and control room dose within the regulatory acceptance criteria.

~~The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip.~~

The analysis assumes

~~With offsite power available,~~ the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the ~~high pressure safety injection (HPSI)~~ pumps, is delayed. Significant single failures considered include: failure of a ~~MSIV~~ to close, failure of a~~n emergency diesel generator~~, and failure of a ~~HPSI pump~~.

containment spray

MFIV

containment cooling train

MSIV to close

to close

1

3

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An HELB inside containment. ~~In order to maximize the mass and energy release into the containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator, and from the residual steam in the main steam header downstream of the closed MSIV in the intact loop~~.

The location of this break is upstream of the MSIVs. The location results in the largest possible steam flow for a given break size. Blowdown of the other steam generator by backflow is prevented by closure of the steam isolation valve in the broken steam line. Blowdown of the intact steam generator is prevented by the steam isolation valve in the unbroken steam line. A MSIS occurs on low steam generator pressure to initiate the closure of the MSIVs.

b. A break outside of containment and upstream from the MSIVs. This scenario is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break, and limits the blowdown to a single steam generator.

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. A break downstream of the MSIVs. This type of break will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the steam bypass valves will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIV~~[~~s~~]~~ isolates the affected steam generator from the intact steam generator. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture. (2)

e. The MSIVs are also utilized during other events such as a feedwater line break. These events are less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that the MSIV in each of the ~~[~~two~~]~~ steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal. (2)

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the ~~10 CFR 100~~ (Ref. 4) limits ~~or the NRC staff approved licensing basis~~. (1)

10 CFR 50.67

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1 and in MODES 2 and 3 except when all MSIVs are closed ~~and [deactivated]~~. In these MODES there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing their safety function. (1) (2)

~~In MODE 4, the steam generator energy is low; therefore, the MSIVs are not required to be OPERABLE.~~

4,

In MODES 5 and 6, the steam generator~~s do not contain much~~ energy ~~because their temperature is below the boiling point of water~~; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES. (1)

is low

BASES

ACTIONS

A.1

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the unit hot. The ~~[~~8~~]~~ hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 2

The ~~[~~8~~]~~ hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation. 2

B.1

If the MSIV cannot be restored to OPERABLE status within ~~[~~8~~]~~ hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2, and close the MSIVs in an orderly manner and without challenging unit systems. 2

C.1~~, C.2.1,~~ and C.2~~.2~~ 4

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The ~~[~~8~~]~~ hour Completion Time is consistent with that allowed in Condition A. 2

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position.

BASES

ACTIONS (continued)

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within ~~[~~12~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is within the limit given in Reference ~~5~~ and is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the INSERVICE TESTING PROGRAM. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. ~~6~~), requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the INSERVICE TESTING PROGRAM.

This test is conducted in MODE 3, with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, in order to establish conditions consistent with those under which the acceptance criterion was generated.

This test is conducted in MODE 3, with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, in order to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. ~~[The Frequency of MSIV testing is every [18] months. The [18] month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

REFERENCES

1. FSAR, Section ~~[~~10.3~~]~~.
2. FSAR, Section ~~[~~6.2~~]~~.
3. FSAR, Section ~~[15.1. 5]~~.
4. ~~10 CFR 100.11~~.
5. ~~[Technical Requirements Manual].~~
6. ASME Code for Operation and Maintenance of Nuclear Power Plants.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.2 BASES, MAIN STEAM ISOLATION VALVES (MSIVs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS Bases contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Editorial/grammatical changes made for enhanced clarity.

4. Changes are made to be consistent with the Specification.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.2, MAIN STEAM ISOLATION VALVES (MSIVs)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

ITS 3.7.3, Main Feedwater Isolation Valves (MFIVs)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3.7 PLANT SYSTEMS

Add proposed ITS 3.7.3

M01

PLANT SYSTEMS

MAIN FEEDWATER ISOLATION VALVES (MFIVs)

LIMITING CONDITION FOR OPERATION

LCO 3.7.3 3.7.1.6 Four ~~main feedwater isolation valves (~~MFIVs~~)~~ shall be OPERABLE.

ACTIONS Note Applicability **APPLICABILITY:*** MODES 1, 2 and 3, except when ~~the~~ MFIV is closed and deactivated.

ACTION:

Action A a. ~~With o~~ne ~~MFIV~~ inoperable ~~in one or more main feedwater lines, OPERATION may continue provided each inoperable valve is restored to OPERABLE status,~~ close~~d,~~ or isolate~~d~~ within 72 hours. Otherwise, be in HOT STANDBY within ~~the next~~ 6 hours and in HOT SHUTDOWN within ~~the following 6~~ hours.

O; or more MFIVs; inoperable MFIV; 12; A02

Action B b. ~~With t~~wo ~~MFIVs inoperable~~ in the same flowpath, ~~restore at least one of the inoperable MFIVs to OPERABLE status or close one of the inoperable valves~~ within ~~4~~ hours. Otherwise, be in HOT STANDBY within ~~the next~~ 6 hours and in HOT SHUTDOWN ~~within the following 6~~ hours.

T; valves; inoperable; isolate affected flow path; 8; 12; A02; L01

Action C

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 4.7.1.6.a ~~Each MFIV shall be demonstrated OPERABLE by v~~erify~~ing full closure within 5.15 seconds when tested pursuant to~~ the INSERVICE TESTING PROGRAM. ~~The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.~~

V; the isolation time of each MFIV is within limits; in accordance with; LA01; M02

Required Actions A.2 and B.2 4.7.1.6.b ~~For each~~ inoperable MFIV~~, verify that it~~ is closed or isolated once per 7 days.

Verify; M03

Add proposed SR 3.7.3.2

ACTIONS Note * ~~Each MFIV shall be treated independently.~~

Separate Condition entry is allowed for each valve.

DISCUSSION OF CHANGES
ITS 3.7.3, MAIN FEEDWATER ISOLATION VALVES (MFIVs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 2 only:** CTS 3.7.1.6 Action a requires the restoration of each inoperable valve to OPERABLE status or close or isolate the inoperable MFIV within 72 hours. CTS 3.7.1.6 Action b requires that at least one of the inoperable MFIVs be restored to OPERABLE status or close one of the inoperable valves within 4 hours. ITS 3.7.3 does not state a requirement to restore an inoperable valve to OPERABLE status but includes other compensatory Required Actions to take within 72 hours or 4 hours, as applicable. This changes the CTS by not explicitly stating the requirement to restore an inoperable valve to OPERABLE status.

This change is acceptable because the technical requirements have not changed. Restoration of compliance with the LCO is always an available Required Action and it is the convention in the ITS to not state such "restore" options explicitly unless it is the only action or is required for clarity. This change is designated as administrative because it does not result in any technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only**: CTS does not have a requirement for Main Feedwater Isolation Valves (MFIVs) to be OPERABLE, Required Actions or Surveillance Requirements.

ITS LCO 3.7.3 requires MFIVs to be OPERABLE in MODES 1, 2 and 3, except when a MFIV is closed and deactivated. In addition, ITS 3.7.3 provides the necessary Required Actions and Surveillance Requirement that support MFIV OPERABILITY. SR 3.7.3.1 will be performed at a Frequency in accordance with the INSERVICE TESTING PROGRAM. SR 3.7.3.2 will be performed at a frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 18 months consistent with the ISTS. In addition, PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per CTS 6.8.4.o. Industry operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. This changes the CTS by incorporating the requirements of ITS 3.7.3.

The safety related function of MFIVs is to provide isolation of the feedwater system from the steam generators and AFW system during normal and emergency conditions. This change is acceptable because the safety analyses

assume the MFIVs function as designed. This change is designated as more restrictive because it adds new requirements to the CTS.

M02 **Unit 2 only:** CTS 4.7.1.6.a states that the provisions of Specification 4.0.4 are not applicable for entry into MODE 3, and thereby provides an allowance for entering the next higher MODE of Applicability when the Surveillance is not met. ITS 3.7.3 does not provide a Surveillance Note that states that the provisions of LCO 4.0.4 are not applicable. LCO 4.0.4 states that entry into a MODE or other specified condition in the Applicability of a Limiting Condition for Operation (LCO) shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by Surveillance Requirement 4.0.3. LCO 4.0.3 provides the provisions for a missed surveillance and does not apply to a surveillance known to not be met within its specified Frequency prior to entering the Mode of Applicability.

The purpose of CTS 4.7.1.6.a is to provide an allowance for entering the MODE of applicability when any Surveillance is not met. This changes CTS by allowing entry into the MODE of Applicability by only deferring the performance of the Surveillance Requirements instead of deferring compliance with the LCO. This change is designated as more restrictive because the CTS 4.0.4 MODE change allowance is deleted and entry into the applicable MODES shall only be made when the LCO's surveillances have been met within their specified Frequency.

M03 **Unit 2 only:** Unit 2 CTS does not require testing to verify that MFIVs close on an actuation signal. ITS SR 3.7.3.2 requires verification that each MFIV actuates to the isolation position on an actual or simulated actuation signal. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.q. Therefore, SR 3.7.3.2 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 18 months consistent with the ISTS SR 3.7.3.2 and considers operating practice to regularly assess potential degradation and verify operation within the safety analysis. This changes the CTS by requiring verification that each MFIV actuates to the isolation position on an actual or simulated actuation signal.

The purpose of the ITS SR 3.7.3.2 is to verify each MFIV can close on an actual or simulated actuation signal. This change is acceptable because the test is conducted to ensure that the MFIV will perform its safety function. The SFCP was established as described in FPL (PSL Unit 1 and Unit 2) "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program" (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066). This change is designated as more restrictive because it adds additional requirements to the CTS.

DISCUSSION OF CHANGES
ITS 3.7.3, MAIN FEEDWATER ISOLATION VALVES (MFIVs)

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 **Unit 2 only:** *(Type 3 - Removing Procedural Detail for Meeting TS Requirements or Reporting Requirements)* CTS 4.7.1.6.a requires verification that the closure time of each MFIV is within 5.15 seconds. ITS SR 3.7.3.1 requires verification that the closure time of each MFIV is within limits. This changes the CTS by moving the MFIV closure time limit to the UFSAR. | R1

The removal of MFIV closure time from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to verify that the closure time of each MFIV is within limits. This change is acceptable because these types of procedural details will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a procedural detail for meeting Technical Specification requirements are being removed from the Technical Specifications. This change is designated as less restrictive, because the ITS LCO requirements are applicable in fewer operating conditions than in the CTS.

LESS RESTRICTIVE CHANGES

L01 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* CTS 3.7.1.6 Action b states that with two MFIVs inoperable in the same flow path restore at least one of the inoperable MFIVs to OPERABLE status or close one of the inoperable valves within 4 hours. ITS 3.7.3 ACTION B when two valves in the same flow path inoperable requires isolating the affected flow path within 8 hours. This changes the CTS 3.7.1.6 time to isolate an affected flow path from 4 hours to 8 hours. | R1

The purpose of the CTS 3.7.1.6 Action is to return the system to the condition where at least one valve in each flow path is performing the required safety function. The CTS action time of 4 hours is consistent with the time required to close an inoperable containment isolation valve. This change is consistent with the ISTS and acceptable because the MFIVs are associated with a containment penetration with a closed system (i.e., steam generator tubes). The relaxed Completion Time is consistent with safe operation under the specified Condition, the capacity and capability of remaining features (i.e., a closed system), a reasonable time for repairs of required features, and the low probability of a DBA occurring during the allowed Completion Time. In the case of two inoperable MFIVs, 8 hours is a reasonable time period considering the relative stability of the closed system to act as a penetration isolation boundary. This change is designated as less restrictive because additional time is allowed to isolate the affected flow path than was allowed in the CTS. | R1

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs) [and [MFIV] Bypass Valves]

DOC M01 LCO 3.7.3 [Two] MFIVs [and [MFIV] bypass valves] shall be OPERABLE. 2

DOC M01 APPLICABILITY: MODES 1, 2, [and 3] except when MFIV [or [MFIV] bypass valve] is closed and [de-activated] or [isolated by a closed manual valve]. 2

ACTIONS

----------NOTE----------

DOC M01 Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01 A. One or more MFIVs [or [MFIV] bypass valves] inoperable.	A.1 Close or isolate inoperable MFIV [or [MFIV] bypass valve].	[8 or 72] hours
	AND	
DOC M01	A.2 Verify inoperable MFIV [or [MFIV] bypass valve] is closed or isolated.	Once per 7 days 2
B. [[Two] valves in the same flow path inoperable.	B.1 Isolate affected flow path.	8 hours
	AND	
	B.2 Verify inoperable MFIV [or [MFIV] bypass valve] is closed or isolated.	Once per 7 days] 3 2
B DOC M01 C. Required Action and associated Completion Time not met.	B C.1 Be in MODE 3.	6 hours 3
	[AND	
	B C. 2 Be in MODE 4.	[12] hours] 2

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
DOC M01	SR 3.7.3.1	Verify the isolation time of each MFIV ~~[and [MFIV] bypass valve]~~ is within limits.	In accordance with the INSERVICE TESTING PROGRAM (2)
DOC M01	SR 3.7.3.2	Verify each MFIV ~~[and [MFIV] bypass valve]~~ actuates to the isolation position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)

CTS

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs) [and [MFIV] Bypass Valves]

3.7.1.6 LCO 3.7.3 Four [Two] MFIVs [and [MFIV] bypass valves] shall be OPERABLE. 2

Applicability APPLICABILITY: MODES 1, 2, [and 3] except when MFIV [or [MFIV] bypass valve] is closed and [de-activated] or [isolated by a closed manual valve].

ACTIONS

----------NOTE----------
Footnote * Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a A. One or more MFIVs [or [MFIV] bypass valves] inoperable.	A.1 Close or isolate inoperable MFIV [or [MFIV] bypass valve]. AND A.2 Verify inoperable MFIV [or [MFIV] bypass valve] is closed or isolated.	[8 or 72] hours Once per 7 days SR 4.7.1.6.b
Action b DOC L01 B. [[Two] valves in the same flow path inoperable.	B.1 Isolate affected flow path. AND B.2 Verify inoperable MFIV [or [MFIV] bypass valve] is closed or isolated.	8 hours Once per 7 days] SR 4.7.1.6.b
Action a Action b C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. [AND C.2 Be in MODE 4.	6 hours [12] hours]

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
SR 4.7.1.6.a DOC LA01 DOC M02	SR 3.7.3.1	Verify the isolation time of each MFIV ~~[and [MFIV] bypass valve]~~ is within limits.	In accordance with the INSERVICE TESTING PROGRAM (2)
DOC M03	SR 3.7.3.2	Verify each MFIV ~~[and [MFIV] bypass valve]~~ actuates to the isolation position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.3, MAIN FEEDWATER ISOLATION VALVES (MFIVs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. **Unit 1 Only:** ISTS ACTION B is bracketed and deleted based on the Unit 1 design that includes only one MFIV in each feedwater line to its respective steam generator. ISTS ACTION C is renumbered in conformance.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs) ~~[and [MFIV] Bypass Valves]~~ 1

BASES

BACKGROUND

The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Closure of the MFIVs ~~and the bypass valves~~ terminates flow to both steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs ~~and bypass valves~~ effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs. 1

The MFIVs ~~and bypass valves~~ isolate the nonsafety related portions from the safety related portion of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loop. 1

One MFIV is located on each ~~A~~FW line, outside, but close to, containment. The MFIVs are located upstream of the AFW injection point so that AFW may be supplied to a steam generator following MFIV closure. The piping volume from the valve to the steam generator must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB. M 1

The MFIVs ~~and its bypass valves~~ close on receipt of a main steam isolation signal (MSIS) generated by ~~either~~ low steam generator pressure ~~or high containment pressure~~. The MSIS also actuates the main steam isolation valves (MSIVs) to close. The MFIVs ~~and bypass valves~~ may also be actuated manually. In addition to the MFIVs ~~and the bypass valves~~, a check valve inside containment is available to isolate the feedwater line penetrating containment, and to ensure that the consequences of events do not exceed the capacity of the containment heat removal systems. 1

and safety injection actuation signal (SIAS) on low-low pressurizer pressure or high containment pressure

~~A description of the~~ MFIV~~s~~ is ~~found~~ in the FSAR, Section ~~[~~10.4.~~7]~~ (Ref. 1). design U 6 discussed 1 2

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs is established by the analysis for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs ~~and their bypass valves~~ may also be relied on to terminate a steam line break for core response analysis ~~and an excess feedwater flow event~~ upon receipt of a MSIS on ~~high~~ low steam generator ~~level~~ pressure. 1

Failure of an MFIV ~~and the bypass valve~~ to close following an SLB, FWLB, or excess feedwater flow event can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event. 1

The MFIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures that the MFIVs ~~and the bypass valves~~ will isolate MFW flow to the steam generators. Following an FWLB or SLB, these valves will also isolate the nonsafety related portions from the safety related portions of the system. This LCO requires that ~~[two]~~ one MFIV~~s [and [MFIV] bypass valves]~~ in each feedwater line be OPERABLE. The MFIVs ~~and the bypass valves~~ are considered OPERABLE when the isolation times are within limits, and ~~are~~ will close~~d~~ on an ~~isolation actuation signal~~ MSIS and SIAS. 1 2 1

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. ~~If an MSIS on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.~~ 2

APPLICABILITY

The MFIVs ~~and the bypass valves~~ must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. 1

In MODES 1, 2, and 3, the MFIV~~[or [MFIV] bypass valves]~~ s are required to be OPERABLE, except when they are closed and deactivated ~~or isolated by a closed manual valve~~, in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and deactivated ~~or isolated by a closed manual valve~~, they are already performing their safety function. 2 1

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs ~~and the bypass valves~~ are normally closed since MFW is not required. 2

BASES

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each value.

A.1 and A.2

With one or more MFIV~~s~~ ~~or the bypass valve~~ inoperable, action must be taken to close or isolate the inoperable valves within ~~[~~8 ~~or 72]~~ hours. When these valves are closed or isolated, they are performing their required safety function (e.g., to isolate the line). (1) (2)

~~For units with only one MFIV per feedwater line:~~ The ~~[~~8~~]~~ hour Completion Time is reasonable to close the MFIV ~~or its bypass valve, which includes performing a controlled unit shutdown to MODE 2~~. (2)

~~The [72] hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.~~ (1)

~~B.1~~

~~If more than one MFIV or [MFIV] bypass valve in the same flow path cannot be restored to OPERABLE status, then there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such is treated the same as a loss of the isolation capability of this flow path. Under these conditions, valves in each flow path must be restored to OPERABLE status, closed, or the flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable to close the MFIV or its bypass valve, or otherwise isolate the affected flow path.~~

~~Inoperable MFIVs and [MFIV] bypass valves that cannot be restored to OPERABLE status within the Completion Time, but are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that these valves are closed or isolated~~. (1)

BASES

ACTIONS (continued)

B

~~C~~.1 and ~~[C~~.2~~]~~ (3) (2)

If the MFIVs ~~and their bypass valves~~ cannot be restored to OPERABLE status, closed, or isolated in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours ~~[~~, and in MODE 4 within ~~[~~12~~]~~ hours~~]~~. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. (1) (2)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV ~~[and [MFIV] bypass valve]~~ is within the limit given in Reference 2 and is within that assumed in the accident and containment analyses. This SR also verifies that the valve closure time is in accordance with the INSERVICE TESTING PROGRAM. This SR is normally performed upon returning the unit to operation following a refueling outage. The MFIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. As these valves are not tested at power, they are exempt from the ASME Code (Ref. 3) requirements during operation in MODES 1 and 2. (2)

The Frequency is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.7.3.2

This SR verifies that each MFIV ~~[and [MFIV] bypass valve]~~ can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. (2)

~~[The Frequency for this SR is every [18] months. The [18] month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.~~ (2)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ (2)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ (4)

REFERENCES

1. FSAR, Section ~~[~~10.4.~~7]~~. (6) (U)
2. ~~[Technical Requirements Manual]~~. (UFSAR, Section 6.2) (1) (2) R1
3. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs) ~~[and [MFIV] Bypass Valves]~~

BASES

BACKGROUND

The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Closure of the MFIVs ~~and the bypass valves~~ terminates flow to both steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs ~~and bypass valves~~ effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

The MFIVs ~~and bypass valves~~ isolate the nonsafety related portions from the safety related portion of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loop.

Two s are M

~~One~~ MFIV ~~is~~ located on each ~~A~~FW line, outside, but close to, containment. The MFIVs are located upstream of the AFW injection point so that AFW may be supplied to a steam generator following MFIV closure. The piping volume from the valve to the steam generator must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

The MFIVs ~~and its bypass valves~~ close on receipt of a main steam isolation signal (MSIS) generated by ~~either~~ low steam generator pressure or high containment pressure. The MSIS also actuates the main steam isolation valves (MSIVs) to close. The MFIVs ~~and bypass valves~~ may also be actuated manually. In addition to the MFIVs ~~and the bypass valves~~, a check valve inside containment is available to isolate the feedwater line penetrating containment, and to ensure that the consequences of events do not exceed the capacity of the containment heat removal systems.

and Auxiliary Feedwater Actuation Signal (AFAS) on low steam generator level

A description of the MFIVs is found in the FSAR, Section ~~[~~10.4.7~~]~~ (Ref. 1).

U

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs is established by the analysis for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs ~~and their bypass valves~~ may also be relied on to terminate a steam line break for core response analysis and an excess feedwater flow event upon receipt of a MSIS on ~~high~~ low steam generator ~~level~~ pressure or high containment pressure. 1

Failure of an MFIV ~~and the bypass valve~~ to close following an SLB, FWLB, or excess feedwater flow event can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event. 1

The MFIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures that the MFIVs ~~and the bypass valves~~ will isolate MFW flow to the steam generators. Following an FWLB or SLB, these valves will also isolate the nonsafety related portions from the safety related portions of the system. This LCO requires that ~~[~~two~~]~~ MFIVs ~~[and [MFIV] bypass valves]~~ in each feedwater line be OPERABLE. The MFIVs ~~and the bypass valves~~ are considered OPERABLE when the isolation times are within limits, and ~~are~~ will close~~d~~ on an ~~isolation actuation signal~~ MSIS and AFAS. 1 2 | R1 1

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. ~~If an MSIS on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.~~ 2

APPLICABILITY

The MFIVs ~~and the bypass valves~~ must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. 1

In MODES 1, 2, and 3, the MFIV~~~~s ~~[or [MFIV] bypass valves]~~ are required to be OPERABLE, except when they are closed and deactivated ~~or isolated by a closed manual valve~~, in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and deactivated ~~or isolated by a closed manual valve~~, they are already performing their safety function. 2 1

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs ~~and the bypass valves~~ are normally closed since MFW is not required. 2

BASES

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each value.

A.1 and A.2

With one or more MFIV~~s~~ ~~or the bypass valve~~ inoperable, action must be taken to close or isolate the inoperable valves within ~~[8 or~~ 72~~]~~ hours. When these valves are closed or isolated, they are performing their required safety function (e.g., to isolate the line). 1 2

~~For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable to close the MFIV or its bypass valve, which includes performing a controlled unit shutdown to MODE 2.~~ 1

The ~~[~~72~~]~~ hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. 2

B.1

If more than one MFIV ~~or [MFIV] bypass valve~~ in the same flow path cannot be restored to OPERABLE status, then there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such is treated the same as a loss of the isolation capability of this flow path. Under these conditions, valves in each flow path must be restored to OPERABLE status, closed, or the flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable to close the MFIV or its bypass valve, or otherwise isolate the affected flow path. 2 1

Inoperable MFIVs ~~and [MFIV] bypass valves~~ that cannot be restored to OPERABLE status within the Completion Time, but are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that these valves are closed or isolated. 1 2

BASES

ACTIONS (continued)

C.1 and ~~[~~C.2~~]~~ (2)

If the MFIVs ~~and their bypass valves~~ cannot be restored to OPERABLE status, closed, or isolated in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours ~~[~~, and in MODE 4 within ~~[~~12~~]~~ hours~~]~~. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. (1) (2)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV ~~[and [MFIV] bypass valve]~~ is within the limit given in Reference 2 and is within that assumed in the accident and containment analyses. This SR also verifies that the valve closure time is in accordance with the INSERVICE TESTING PROGRAM. This SR is normally performed upon returning the unit to operation following a refueling outage. The MFIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. As these valves are not tested at power, they are exempt from the ASME Code (Ref. 3) requirements during operation in MODES 1 and 2. (2)

The Frequency is in accordance with the INSERVICE TESTING PROGRAM.

SR 3.7.3.2

This SR verifies that each MFIV ~~[and [MFIV] bypass valve]~~ can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. (2)

~~[The Frequency for this SR is every [18] months. The [18] month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.~~ (2)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4

REFERENCES

1. FSAR, Section ~~[~~10.4.7~~]~~. U
2. ~~[Technical Requirements Manual]~~. UFSAR, Section 6.2
3. ASME Code for Operation and Maintenance of Nuclear Power Plants.

1 2 R1

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.3 BASES, MAIN FEEDWATER ISOLATION VALVES (MFIVs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS Bases contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made to be consistent with the Specification.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.3, MAIN FEEDWATER ISOLATION VALVES (MFIVs)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 4

ITS 3.7.4, Atmospheric Dump Valves (ADVs)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3.7 PLANT SYSTEMS

Add proposed ITS 3.7.4

M01

PLANT SYSTEMS

ATMOSPHERIC DUMP VALVES

(ADVs)

LIMITING CONDITION FOR OPERATION

LCO 3.7.4

3.7.1.7 Two ADV lines ~~The atmospheric dump and associated block valves~~ shall be OPERABLE ~~with:~~ LA01

~~a. All atmospheric dump valves in manual control above 15% of RATED THERMAL POWER, and~~

~~b. No more than one atmospheric dump valve per steam generator in automatic control below 15% of RATED THERMAL POWER~~. L01

Applicability

APPLICABILITY: MODE 1~~:~~ , 2, and 3, M02

Add proposed second Applicability

ACTION:

Action A

a. One ADV line inoperable ~~With less than one atmospheric dump and associated block valve per steam generator OPERABLE,~~ restore ~~the required atmospheric dump and associated block valve~~ to OPERABLE status within 72 hours, or ADV line LA01

Action C

be in ~~at least~~ HOT STANDBY within ~~the next~~ 6 hours~~.~~ AND be in MODE 4 without reliance upon steam generator for heat removal within 24 hours M03

~~b. With more than the permissible number of atmospheric dump valves in automatic control, return the atmospheric dump valves to manual control within 1 hour, or be in at least HOT STANDBY within the next 6 hours.~~ L01

Add proposed ACTION B M04

SURVEILLANCE REQUIREMENTS

~~4.7.1.7 Each atmospheric dump valve shall be verified to be in the manual operation mode in accordance with the Surveillance Frequency Control Program during operation at ≥ 15% of RATED THERMAL POWER.~~ L01

Add proposed SR 3.7.4.1

Add proposed SR 3.7.4.2 M05

DISCUSSION OF CHANGES
ITS 3.7.4, ATMOSPHERIC DUMP VALVES (ADVs)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only**: CTS does not have a Specification for Atmospheric Dump Valves (ADVs). ITS LCO 3.7.4 requires ADV lines to be OPERABLE in MODES 1, 2, 3, and MODE 4 when the steam generator is being relied upon for heat removal. In addition, ITS 3.7.4 provides the necessary Required Actions to provide appropriate remedial actions when the LCO is not met and Surveillance Requirement that support ADV line OPERABILITY. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per CTS 6.8.4.o. Therefore, SR 3.7.4.1 and SR 3.7.4.2 will be performed at a frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 18 months consistent with the ISTS. This changes the CTS by incorporating requirements for ADVs in ITS 3.7.4.

The safety related function of ADVs is to provide the capability to cool the unit to Shutdown Cooling (SDC) System entry conditions. The ADVs are assumed to be used by the operator to cool down the unit to SDC entry conditions for accidents accompanied by a loss of offsite power. This change is necessary because the safety analyses assume the ADVs function, coincident with a loss of offsite power, in the main steam line break upstream of the main steam isolation valves, a feedwater line break, and a steam generator tube rupture event. This change is designated as more restrictive because it adds new requirements to the CTS.

M02 **Unit 2 only:** CTS 3.7.1.7 is applicable in MODE 1. ITS 3.7.4 Applicability states MODES 1, 2, and 3, and MODE 4 when steam generator is being relied upon for heat removal. This changes the CTS to require OPERABILITY of the ADV lines, which include ADVs and associated block valves, in additional MODES of operation. The purpose of CTS 3.7.1.7 is to ensure the capability to cool the unit to SDC System entry conditions. These changes are acceptable because they ensure the necessary decay heat removal capacity is available when a steam generator is relied upon for heat removal to support the safety analysis and licensing basis.

These changes are designated as more restrictive because they add additional Applicability requirements that are not required by the CTS.

M03 **Unit 2 only:** CTS 3.7.1.7 Action a states, in part, to be in at least HOT STANDBY (MODE 3) within the next 6 hours. ITS 3.7.4 ACTION C requires the unit to be in

MODE 3 within 6 hours and in MODE 4 without reliance upon steam generator for heat removal within 24 hours. This changes the CTS by adding an additional requirement to be in MODE 4 without reliance upon steam generator for heat removal within 24 hours.

The purpose of the shutdown Actions is to place the unit outside of the Applicability of the Specification. ITS 3.7.4 ACTION C accomplishes this purpose by adding the additional requirement. (See DOC M02 for discussion of changes to the Applicability.) This change is acceptable because the requirement to be in MODE 4 without reliance upon steam generator for heat removal places the unit outside of the Applicability and the time required to be in MODE 4 is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. This change is designated as more restrictive because additional Required Actions are now required that were not required in CTS.

M04 **Unit 2 only:** CTS 3.7.1.7 action a allows 72 hours to restore the required atmospheric dump and associated block valve to OPERABLE status with less than one ADV and associated block valve per steam generator OPERABLE. ITS 3.7.4 ACTION B requires the restoration of at least one ADV line to OPERABLE status within 24 hours whenever two ADV lines are inoperable. This changes CTS by adding an additional Required Action and Completion time when two ADV lines are inoperable.

The purpose of ITS 3.7.4 ACTION B is to restore one of the ADV lines to OPERABLE status when both ADV lines are inoperable (e.g., one or more ADVs or one or more block valves in both ADV lines are inoperable). Because the block valve can be closed to isolate an ADV line, repair may be possible with the unit at power and the 24 hour Completion Time is reasonable based on the availability of the steam bypass system and MSSVs, and the low probability of an event occurring during the period requiring the ADV lines. This change is designated as more restrictive because additional requirements are being added to the ITS than are required by the CTS.

M05 **Unit 2 only:** CTS LCO 3.7.1.7 requires the atmospheric dump and associated block valves shall be OPERABLE. The CTS does not include any Surveillance to verify the ADVs and associated block valves are capable of performing their safety function (i.e., manually open to cooldown the unit during accidents with a loss of offsite power and close to preclude an uncontrolled steam release). ITS SR 3.7.4.1 requires verification of one complete cycle of each ADV and SR 3.7.4.2 requires verification of one complete cycle of each ADV block valve. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.q. Therefore, ITS SRs 3.7.4.1 and 3.7.4.2 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 18 months consistent with the ISTS and considers industry operating experience that has shown that components of this type usually pass the SR when performed at an 18 month Frequency. This changes the CTS by adding SRs to verify one complete cycle of each ADV and associated block valve. | R1

The purpose of ITS 3.7.4 is to ensure a safety grade method for cooling the unit to shutdown cooling entry conditions is available should the preferred method, via the steam bypass system to the condenser, not be available. This change is necessary because the ADV line is required to be OPERABLE to ensure the unit can be cooled to shutdown cooling entry conditions in the event of one of the steam generators becomes unavailable coincident with a loss of offsite power. SR 3.7.4.1 is being added to verify the function of the ADV, which is to be opened and throttled through its full range. Cycling the ADV closed and open demonstrates its capability to perform this function. SR 3.7.4.2 is being added to verify the function of the ADV block valve, which is to isolate a failed open ADV. Cycling the block valve closed and open demonstrates its capability to perform this function. The SFCP was established as described in FPL (PSL Unit 1 and Unit 2) "Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program" (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066). This change is designated as more restrictive because additional requirements are being added to the ITS than are required by the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 **Unit 2 only:** *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* CTS 3.7.1.7 requires, in part, the atmospheric dump and associated block valves to be OPERABLE. CTS 3.7.1.7 Action a requires, with less than one atmospheric dump and associated block valve per steam generator OPERABLE, the required atmospheric dump and associated block valve be restored to OPERABLE status within 72 hours. ITS LCO 3.7.4 requires two ADV lines to be OPERABLE. ITS 3.7.4 ACTION A requires, with one ADV line inoperable, restoration of the inoperable ADV line to OPERABLE status within 72 hours. ITS 3.7.4 does not include the detail of what constitutes an OPERABLE ADV line. This changes the CTS by moving the details of what constitutes an OPERABLE ADV line to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the ADVs and associated block valves (i.e., ADV line) be OPERABLE and provides sufficient remedial actions to restore these components to OPERABLE status. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail

change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 **Unit 2 only:** *(Category 1 – Relaxation of LCO Requirements)* CTS 3.7.1.7 requires that the atmospheric dump and associated block valves shall be OPERABLE with: a) all atmospheric dump valves in manual control above 15% of RATED THERMAL POWER; and b) no more than one atmospheric dump valve per steam generator in automatic control below 15% of RATED THERMAL POWER. CTS 3.7.1.7 Action b and CTS 4.7.1.7 also provide requirements in support of CTS LCO 3.7.1.7.a and b. ITS 3.7.4 requires that two ADV lines shall be OPERABLE. Refer to DOC LA01 for additional description of changes. This changes the CTS by deleting procedural operating detail from the Technical Specifications, including CTS 3.7.1.7 Action b and CTS 4.7.1.7.

The purpose of CTS 3.7.1.7 is to ensure that sufficient ADV capacity is available for an operator to conduct a unit cooldown following an event with a loss of offsite power. This change is acceptable because the LCO and Surveillance requirements continue to ensure that the ADV capability is maintained consistent with the safety analysis and licensing basis. To support a unit cooldown during an accident concurrent with a loss of offsite power, the operator manually opens the ADV and, if necessary, the associated block valve. This can be accomplished whether the ADV controller is in automatic or manual. Additionally, if the ADV spuriously opens while THERMAL POWER is above 15% RTP, an OPERABLE block valve can be closed to isolate the uncontrolled steam release. The deleted procedural detail is a prudent operational consideration and more appropriately addressed and controlled in plant operating procedures. This change is designated as less restrictive because less stringent LCO requirements, Actions, and Surveillance Requirements are being applied in ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valves (ADVs)

DOC M01 LCO 3.7.4 ~~[~~Two~~]~~ ADV lines shall be OPERABLE. (2)

DOC M01 APPLICABILITY: MODES 1, 2, and 3,
~~[~~MODE 4 when steam generator is being relied upon for heat removal~~]~~. (2)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01 ~~A. One required ADV line inoperable.~~	~~A.1 Restore ADV line to OPERABLE status.~~	~~7 days~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ (1)
DOC M01 A → ~~B~~. ~~Two~~ One or more ~~[required]~~ ADV lines inoperable.	~~B~~.1 A → Restore all ~~but one~~ ADV line(s) to OPERABLE status.	24 hours (1)(2)
DOC M01 B → ~~C~~. Required Action and associated Completion Time not met.	~~C~~.1 B → Be in MODE 3. ~~[~~ AND ~~C~~.2 B → Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours (1) ~~[~~24~~]~~ hours~~]~~ (2)

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
DOC M01	SR 3.7.4.1	Verify one complete cycle of each ADV.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)
DOC M01	SR 3.7.4.2	~~[~~ Verify one complete cycle of each ADV block valve.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~ (2)

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valves (ADVs)

3.7.1.7 DOC LA01 DOC L01

LCO 3.7.4 [Two] ADV lines shall be OPERABLE. (2)

Applicability DOC M02

APPLICABILITY: MODES 1, 2, and 3,
[MODE 4 when steam generator is being relied upon for heat removal]. (2)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a DOC LA01 A. One ~~required~~ ADV line inoperable.	A.1 Restore ADV line to OPERABLE status.	72 hours ~~7 days~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ (1) (2)
Action b DOC M04 B. Two ~~or more [required]~~ ADV lines inoperable.	B.1 Restore ~~all but~~ one ADV line to OPERABLE status.	24 hours (1) (2)
Actions a and b DOC M03 C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. ~~[~~ AND C.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours ~~[~~24~~]~~ hours~~]~~ (2)

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
DOC M05	SR 3.7.4.1	Verify one complete cycle of each ADV.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)
DOC M05	SR 3.7.4.2	~~[~~ Verify one complete cycle of each ADV block valve.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~ (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.4, ATMOSPHERIC DUMP VALVES (ADVs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a ~~safety grade~~ method for cooling the unit to Shutdown Cooling (SDC) System entry conditions, should the preferred heat sink via the Steam Bypass System to the condenser not be available, as discussed in the FSAR, Section ~~[~~10.3~~]~~ (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Bypass System.

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1

1 2

Two

~~Four~~ ADV lines are provided. Each ADV line consists of one ADV and an associated block valve. Two ADV lines per steam generator are required to meet ~~single failure~~ assumptions following an event ~~rendering one steam generator unavailable~~ for Reactor Coolant System (RCS) heat removal.

, one

in the safety analysis

,

1

The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs have a non-safety automatic pressure control capability. However, the only function of the ADVs required by the safety analyses is the ability to cool down the plant following a design basis event.

~~The ADVs are usually provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ADVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the ADVs for the time required for RCS cooldown to the SDC System entry conditions~~.

air operated valves designed to fail closed on loss of instrument air. The valves are operated from the control room.

A description of the ADVs is found in Reference 1. The ADVs are ~~OPERABLE with only a DC power source available~~. In addition, hand wheels are provided for local manual operation.

1

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to SDC System entry conditions. A cooldown rate of 75°F per hour is obtainable by ~~one or~~ both steam generators. This design is adequate to cool the unit to SDC System entry conditions ~~with only one ADV and one steam generator,~~ utilizing the cooling water supply available in the CST.

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

U

In the accident analysis presented in the FSAR, the ADVs are assumed to be used by the operator to cool down the unit to SDC System entry conditions for accidents accompanied by a loss of offsite power. Prior to the operator action, the main steam safety valves (MSSVs) are used to maintain steam generator pressure and temperature at the MSSV setpoint. ~~This is typically 30 minutes following the initiation of an event. (This may be less for a steam generator tube rupture (SGTR) event.)~~ The limiting events are those that render one steam generator unavailable for RCS heat removal, with a coincident loss of offsite power; this results from a turbine trip and the single failure of one ADV on the unaffected steam generator. Typical initiating events falling into this category are a main steam line break (MSLB) upstream of the main steam isolation valves, a feedwater line break (MFLB), and ~~an SGTR~~ a steam generator tube rupture (SGTR) event ~~(although the ADVs on the affected steam generator may still be available following a SGTR event)~~. 1

The ADVs are not designed to meet redundancy requirements (each steam generator is provided with a single ADV). The short term consequences of design basis events that result in the loss of ADV capability (e.g., SGTR) are not bounded. Manual (local) operation is provided to address long term consequences.

~~The design must accommodate the single failure of one ADV to open on demand; thus, each steam generator must have at least two ADVs.~~ The ADVs are equipped with block valves in the event an ADV spuriously opens, or fails to close during use.

The ADVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii), each with an OPERABLE ADV and an OPERABLE block valve. This ensures

LCO

two ADVs are

~~[~~Two~~]~~ ADV lines are required to be OPERABLE ~~on each steam generator to ensure~~ that at least ~~one ADV is~~ OPERABLE to conduct a unit cooldown following an event ~~in which one steam generator becomes unavailable, accompanied by a single active failure of one ADV line on the unaffected steam generator~~. The block valves must be OPERABLE to isolate a failed open ADV. A closed block valve does not render it or its ADV line inoperable if operator action time to open the block valve is supported in the accident analysis. 2 1

Failure to meet the LCO can result in the inability to cool the unit to SDC System entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An ADV is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and is capable of fully opening and closing on demand.

BASES

APPLICABILITY

In MODES 1, 2, and 3, ~~[~~and in MODE 4, when steam generator is being relied upon for heat removal,~~]~~ the ADVs are required to be OPERABLE.

MSLB, MFLB or — are — s

In MODES 5 and 6, an SGTR ~~is~~ not ~~a~~ credible event.

ACTIONS

A.1

~~With one required ADV line inoperable, action must be taken to restore the OPERABLE status within 7 days [or in accordance with the Risk Informed Completion Time Program]. The 7 day Completion Time takes into account the redundant capability afforded by the remaining OPERABLE ADV lines, and a nonsafety grade backup in the Steam Bypass System and MSSVs.~~

1

~~B.1~~

3

one

With ~~[two]~~ or more ~~[required]~~ ADV lines inoperable, action must be taken to restore ~~[one] of~~ the ADV lines to OPERABLE status. As the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that requires the ADV lines.

2

B

~~C~~.1 and ~~C~~.2

3

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon the steam generator for heat removal, within ~~[~~24~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

2

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened and throttled through their full range. This SR ensures the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. ~~[Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 2

[SR 3.7.4.2

The function of the ADV block valve is to isolate a failed open ADV. Cycling the block valve closed and open demonstrates its capability to perform this function. ~~Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. [Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 1 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]]~~ 4 2

REFERENCES U 1. FSAR, Section ~~[~~10.3~~]~~. 1 2

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a safety grade method for cooling the unit to Shutdown Cooling (SDC) System entry conditions, should the preferred heat sink via the Steam Bypass System to the condenser not be available, as discussed in the FSAR, Section [10.3] (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Bypass System.

U

1 2

Two

two 50% capacity, AC/DC motor operated

s

s

Four ADV lines are provided. Each ADV line consists of one ADV and an associated block valve. Two ADV lines per steam generator are required to meet single failure assumptions following an event rendering one steam generator unavailable for Reactor Coolant System (RCS) heat removal.

line

1

or a loss of one DC electrical power distribution train concurrent with a loss of offsite power

The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

1

capable of automatic modulation using AC power and open and close capability from the control room using AC or DC power only.

The ADVs are usually provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ADVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the ADVs for the time required for RCS cooldown to the SDC System entry conditions.

1

A description of the ADVs is found in Reference 1. The ADVs are OPERABLE with only a DC power source available. In addition, hand wheels are provided for local manual operation.

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to SDC System entry conditions. A cooldown rate of 75°F per hour is obtainable by one or both steam generators. This design is adequate to cool the unit to SDC System entry conditions with only one ADV and one steam generator, utilizing the cooling water supply available in the CST.

two

s

1

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

U

In the accident analysis presented in the FSAR, the ADVs are assumed to be used by the operator to cool down the unit to SDC System entry conditions for accidents accompanied by a loss of offsite power. Prior to the operator action, the main steam safety valves (MSSVs) are used to maintain steam generator pressure and temperature at the MSSV setpoint. ~~This is typically 30 minutes following the initiation of an event. (This may be less for a steam generator tube rupture (SGTR) event.)~~ The limiting event~~s are those that render one steam generator unavailable for RCS heat removal~~, with a coincident loss of offsite power~~; this results from a turbine trip and the single failure of one ADV on the unaffected steam generator~~. Typical initiating events ~~falling into this category~~ are a main steam line break upstream of the main steam isolation valves, a feedwater line break, and ~~an~~ SGTR event ~~(although the ADVs on the affected steam generator may still be available following a SGTR event)~~.

is the loss of one DC emergency train

This event results in a loss of two ADVs, one in each ADV line. The remaining ADV in each line is sufficient to support the capability to cool the unit to SDC entry conditions

a steam generator tube rupture (

)

that results in the failure of two ADVs, one in each ADV line,

or failure of one ADV line to support single generator cooldown

1

The design must accommodate the single failure ~~of one ADV~~ to open on demand; thus, each steam generator must have at least two ADVs. The ADVs are equipped with block valves in the event an ADV spuriously opens, or fails to close during use.

The ADVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

, each with two OPERABLE ADVs and two OPERABLE block valves in each line. This ensures

LCO

two ADVs are

~~[~~Two~~]~~ ADV lines are required to be OPERABLE ~~on each steam generator to ensure~~ that at least ~~one~~ ADV ~~is~~ OPERABLE to conduct a unit cooldown following an event ~~in which one steam generator becomes unavailable~~, accompanied by a single ~~active~~ failure ~~of one ADV line on the unaffected steam generator~~. The block valves must be OPERABLE to isolate a failed open ADV. A closed block valve does not render it or its ADV line inoperable if operator action time to open the block valve is supported in the accident analysis.

2

1

Failure to meet the LCO can result in the inability to cool the unit to SDC System entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An ADV is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and is capable of fully opening and closing on demand.

BASES

APPLICABILITY

In MODES 1, 2, and 3, ~~[~~and in MODE 4, when steam generator is being relied upon for heat removal,~~]~~ the ADVs are required to be OPERABLE. 2

main steam line break, main feedwater line break or

are

s

In MODES 5 and 6, an SGTR ~~is~~ not ~~a~~ credible event. 1

ACTIONS

A.1

(e.g., one or more ADVs or one or more block valves inoperable)

With one ~~required~~ ADV line inoperable, action must be taken to restore the OPERABLE status within ~~7 days [or in accordance with the Risk Informed Completion Time Program]~~. The ~~7 day~~ Completion Time takes into account the redundant capability afforded by the remaining OPERABLE ADV line~~s~~, and a nonsafety grade backup in the Steam Bypass System and MSSVs. 2 1

72 hours

line to

72 hour

B.1

(e.g., two ADVs or two block valves inoperable on each steam generator)

With ~~[~~two~~]~~ ~~or more [required]~~ ADV lines inoperable, action must be taken to restore ~~[~~one~~]~~ of the ADV lines to OPERABLE status. As the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that requires the ADV lines. 2 1

C.1 and C.2

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon the steam generator for heat removal, within ~~[~~24~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. 2

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened and throttled through their full range. This SR ensures the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. ~~[Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4

~~[~~ SR 3.7.4.2

The function of the ADV block valve is to isolate a failed open ADV. Cycling the block valve closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. ~~[Operating experience has shown that these components usually pass the SR when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]]~~ 4 2

REFERENCES U 1. FSAR, Section ~~[~~10.3~~]~~. 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.4 BASES, ATMOSPHERIC DUMP VALVES (ADVs)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS Bases contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made to be consistent with the Specification.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.4, ATMOSPHERIC DUMP VALVES (ADVs)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

ITS 3.7.5, Auxiliary Feedwater (AFW) System

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

PLANT SYSTEMS

(AFW)

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.5

3.7.1.2 ~~At least~~ three ~~independent~~ steam generator auxiliary feedwater ~~pumps and associated flow paths~~ shall be OPERABLE ~~with:~~

T

trains

LA01

M01

LA01

Add proposed LCO Note

~~a.~~ ~~Two motor driven feedwater pumps, and~~

~~b.~~ ~~One feedwater pump capable of being powered from an OPERABLE steam supply system~~.

Applicability

APPLICABILITY: MODES 1, 2 and 3~~.~~

,

Add proposed second Applicability

affected equipment

M01

L01

LA01

ACTION:

Turbine driven AFW train inoperable due to one

Add proposed Action A Note and second Condition

a. ~~With one auxiliary feedwater pump~~ steam supply inoperable, restore ~~the inoperable auxiliary feedwater pump steam supply~~ to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 6~~ hours.

Action A

Action D

MODE 3

12

MODE 4

in MODE 1, 2, or 3 for reasons other than Condition A

train

train

L02

LA01

b. With one auxiliary feedwater ~~pump~~ inoperable, restore the auxiliary feedwater ~~pump~~ to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 6~~ hours.

Action B

Action D

MODE 3

MODE 4

12

Turbine driven AFW train inoperable due to one

train

LA01

c. ~~With one auxiliary feedwater pump~~ steam supply inoperable and one motor-driven auxiliary feedwater ~~pump~~ inoperable, either restore the inoperable auxiliary feedwater ~~pump~~ steam supply OR restore the inoperable motor-driven auxiliary feedwater ~~pump~~ to OPERABLE status within 24 hours or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 6~~ hours.

Action C

Action D

MODE 3

12

MODE 4

in MODE 1, 2, or 3 for reasons other than Condition C

trains

LA01

d. With two auxiliary feedwater ~~pumps~~ inoperable, be in ~~at least HOT STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 6~~ hours.

Action D

MODE 4

12

MODE 3

> **NOTE**
> LCO 3.0.3 and all other LCO Actions requiring MODE changes are suspended until one AFW ~~pump~~ is restored to OPERABLE status.

Action E Note

train

trains

in MODE 1, 2, or 3

LA01

e. With three auxiliary feedwater ~~pumps~~ inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater ~~pump~~ to OPERABLE status.

Action E

train

Actions NOTE

f. LCO 3.0.4.b is not applicable.

Add proposed Action F

M01

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater ~~pump~~ shall be demonstrated OPERABLE:

train

SR 3.7.5.1

a. In accordance with the Surveillance Frequency Control Program by:

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.7.5.1

1. Verify~~ing that~~ each ~~valve (~~manual, power operated ~~or~~ automatic~~)~~ valve in ~~the flow path~~ that is not locked, sealed, or otherwise secured in position, is in its correct position.

and

each water flow path and in both steam flow paths to the steam turbine driven pump,

b. In accordance with the Surveillance Frequency Control Program during shutdown by:

L03 M01

SR 3.7.5.3

1. Verify~~ing that~~ each automatic valve ~~in the flowpath~~ actuates to its correct position ~~upon receipt of the Auto Start actuation test signal~~.

Add proposed SR 3.7.5.3 Notes 1 and 2

that is not locked, sealed, or otherwise secured in position,

L04

on an actual or simulated actuation signal

Add proposed SR 3.7.5.4 Notes 1 and 2

L03 M01

SR 3.7.5.4

2. Verify~~ing that~~ each auxiliary feedwater pump starts automatically ~~as designed upon receipt of the Auto Start actuation test signal~~.

on an actual or simulated actuation signal when in MODE 1, 2, or 3

L05

V

SR 3.7.5.2

c. ~~By v~~erify~~ing~~ the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head when tested in accordance with the INSERVICE TESTING PROGRAM. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 when testing the steam turbine-driven AFW pump and this Surveillance must be performed within 24 hours after entering MODE 3~~ and prior to entering MODE 2~~.

M01

SR 3.7.5.2 Note

L03

M02

Add proposed SR 3.7.5.5

PLANT SYSTEMS

(AFW)

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.5

3.7.1.2 ~~At least~~ T three ~~independent~~ steam generator auxiliary feedwater ~~pumps and associated flow paths~~ trains shall be OPERABLE ~~with:~~ LA01

Add proposed LCO Note

M01

~~a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and~~

LA01

~~b. One feedwater pump capable of being powered from an OPERABLE steam supply system~~.

Applicability **APPLICABILITY**: MODES 1, 2, and 3~~*~~ , Add proposed second Applicability

affected equipment

M01

ACTION:

L01

Turbine driven AFW train inoperable due to one

Add proposed Action A Note and second Condition

LA01

Action A a. ~~With one auxiliary feedwater pump~~ steam supply inoperable, restore ~~the inoperable auxiliary feedwater pump steam supply~~ to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 6~~ hours.

Action D — MODE 3 — 12 — MODE 4

train — train — in MODE 1, 2, or 3 for reasons other than Condition A

L02

LA01

Action B b. With one auxiliary feedwater ~~pump~~ inoperable, restore the auxiliary feedwater ~~pump~~ to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 6~~ hours.

Action D — MODE 3 — 12 — MODE 4

Turbine driven AFW train inoperable due to one

train

Action C c. ~~With one auxiliary feedwater pump~~ steam supply inoperable and one motor-driven auxiliary feedwater ~~pump~~ inoperable, either restore the inoperable auxiliary feedwater ~~pump~~ steam supply OR restore the inoperable motor-driven auxiliary feedwater ~~pump~~ to OPERABLE status within 24 hours or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 6~~ hours.

LA01

Action D — MODE 3 — 12 — MODE 4

in MODE 1, 2, or 3 for reasons other than Condition C

Action D d. With two auxiliary feedwater ~~pumps~~ inoperable, be in ~~at least HOT STANDBY~~ within 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following 6~~ hours.

trains — MODE 3 — MODE 4 — 12

LA01

Action E Note

> **NOTE**
> LCO 3.0.3 and all other LCO Actions requiring MODE changes are suspended until one AFW ~~pump~~ is restored to OPERABLE status.

train

Action E e. With three auxiliary feedwater ~~pumps~~ inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater ~~pump~~ to OPERABLE status.

trains — in MODE 1, 2, or 3 — train

LA01

Actions NOTE f. LCO 3.0.4.b is not applicable.

Add proposed Action F

M01

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater ~~pump~~ shall be demonstrated OPERABLE:

train

a. In accordance with the Surveillance Frequency Control Program by:

SR 3.7.5.1

1. Verify~~ing that~~ each ~~valve (~~manual, power-operated, ~~or~~ automatic~~)~~ in ~~the flow path~~ that is not locked, sealed, or otherwise secured in position, is in its correct position.

and — valve

each water flow path and in both steam flow paths to the steam turbine driven pump,

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program during shutdown by:

L03 M01 L04

Add proposed SR 3.7.5.3 Notes 1 and 2

that is not locked, sealed, or otherwise secured in position,

SR 3.7.5.3

1. Verify~~ing that~~ each automatic valve ~~in the flowpath path~~ actuates to its correct position ~~upon receipt of an auxiliary feedwater actuation test signal~~.

on an actual or simulated actuation signal

Add proposed SR 3.7.5.4 Notes 1 and 2

L03 M01 L05

SR 3.7.5.4

2. Verify~~ing that~~ each pump starts automatically ~~upon receipt of an auxiliary feedwater actuation test signal~~.

on an actual or simulated actuation signal when in MODE 1, 2, or 3

SR 3.7.5.5

c. Following an extended cold shutdown (30 days or longer) and prior to entering MODE 2, a flow test shall be performed to verify the normal flow path from the condensate storage tank (CST) to the steam generators.

V

SR 3.7.5.2

d. ~~By v~~erify~~ing~~ the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head when tested in accordance with the INSERVICE TESTING PROGRAM.

M01

SR 3.7.5.2 Note

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 when testing the steam turbine-driven AFW pump and this Surveillance must be performed within 24 hours after entering MODE 3 ~~and prior to entering MODE 2~~.

L03

DISCUSSION OF CHANGES
ITS 3.7.5, AUXILIARY FEEDWATER (AFW) SYSTEM

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS LCO 3.7.1.2 is applicable in MODES 1, 2, and 3. ITS LCO 3.7.5 is applicable in MODES 1, 2, and 3, and MODE 4 when the steam generator is relied upon for heat removal. To support this change in the Applicability, the following additional requirements are added to the CTS:

- A Note is added to the LCO that requires only one AFW train, which includes a motor driven pump, to be OPERABLE in MODE 4;
- A new ACTION F is added which requires immediate action to restore a required inoperable AFW train to OPERABLE status when the steam generator (SG) is relied upon for heat removal in MODE 4; and
- CTS 4.7.1.2.a, b, and c, which are applicable in MODES 1, 2, and 3, are now applicable in MODE 4 when the SG is relied upon for heat removal (ITS SR 3.7.5.1 and SR 3.7.5.2) for the required AFW train.

These changes are acceptable because they ensure the necessary support systems are available when a steam generator is being relied upon for heat removal in MODE 4. The CTS does not have specific requirements for an AFW train to be OPERABLE in MODE 4 when a steam generator is relied upon for heat removal. One AFW train, supplied by a motor driven pump, will provide sufficient water to the SG to remove decay heat in MODE 4. If the required AFW train is inoperable, ITS 3.7.5 ACTION F requires the initiation of action to restore the AFW train to OPERABLE status immediately. ITS SR 3.7.5.1 and SR 3.7.5.2 ensure the required AFW train is OPERABLE. This is acceptable because without the SG it may not be possible to cool down the unit and exit the MODE of Applicability. A modifying Note (Note 2) is added to ITS SR 3.7.5.3 (CTS 4.7.1.2.b.1) and ITS SR 3.7.5.4 (CTS 4.7.1.2.b.2) that state the SRs are not required to be met while in MODE 4 when the steam generator is relied upon for heat removal. This is acceptable because in MODE 4 the heat removal requirements are less providing more time for operator action to manually start a required AFW pump or the required AFW train is already aligned and operating. These changes are designated as more restrictive because they place additional requirements on unit operations in MODE 4 that are not required by the CTS.

M02 **Unit 1 only:** CTS 3.7.1.2 does not provide a surveillance requirement to verify proper alignment of the required AFW flowpaths prior to entering MODE 2 when the unit has been in MODE 5, MODE 6, or defueled for a cumulative period greater than 30 days. ITS SR 3.7.5.5 requires verification of the proper

alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator prior to entering MODE 2 whenever the unit has been in MODE 5, MODE 6, or defueled for a cumulative period of greater than 30 days. This changes CTS 3.7.1.2 by adding a specific surveillance requirement (ITS SR 3.7.5.5) to verify alignment of the required AFW flow paths from the condensate storage tank to each steam generator prior to entering MODE 2 when the unit has been in MODE 5, MODE 6, or defueled for a cumulative period of greater than 30 days.

The purpose of ITS 3.7.5.5 is to verify that the AFW system is properly aligned by verifying the flow path to each steam generator prior to entering MODE 2 operation after 30 days in any combination of MODE 5 or 6, or defueled. This change is acceptable because it verifies the flow path from the CST to the steam generators is properly aligned by requiring verification that the minimum flow capacity is satisfied. This change is designated as more restrictive because ITS 3.7.5.5 contains surveillance requirements not previously specified in CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS LCO 3.7.1.2 requires three “independent” steam generator AFW “pumps and associated flow paths” to be OPERABLE. This includes two motor driven AFW pumps powered from separate emergency buses, and the steam turbine driven AFW pump capable of being powered from an OPERABLE steam supply system. ITS LCO 3.7.5 states “Three AFW trains shall be OPERABLE.” The ITS does not include design details or define the components and associated flow paths that comprise an OPERABLE AFW train. CTS 3.7.1.2 Actions a, b, c, d and e cover the inoperabilities associated with the auxiliary feedwater pump(s). ITS 3.7.5 ACTIONS A, B, C, D, and E cover the inoperabilities of the train(s) which includes both the pump and the associated flow path. This changes the CTS by moving the description of the AFW System to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included to provide adequate protection of public health and safety. The ITS retains all necessary requirements in the LCO to ensure OPERABILITY for the AFW trains. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.7.1.2 Action a requires with one auxiliary feedwater pump steam supply inoperable to be restored to an OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program. CTS 3.7.1.2 Action b requires an inoperable AFW pump to be restored to an OPERABLE status within 72 hours for any condition of inoperability. ITS 3.7.5 ACTION A permits 7 days to restore the affected equipment to OPERABLE status when the turbine driven AFW pump is inoperable due to an inoperable steam supply valve or if the turbine driven AFW pump is inoperable in MODE 3 following refueling. In addition, the Condition is modified by a Note that restricts the Condition to "only applicable if MODE 2 has not been entered following refueling." This changes the CTS by adding the Condition of the turbine driven AFW pump inoperable in MODE 3 following refueling with restoration of the affected equipment to OPERABLE status required within 7 days or in accordance with the Risk Informed Completion Time and restricts the applicability of the condition to only if MODE 2 has not been entered.

The purpose of CTS 3.7.1.2 Action a is to provide a limit on the length of time the unit may remain in the MODES of Applicability with the turbine driven AFW pump inoperable. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions to minimize risk associated with continued operation while providing time to repair the inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. An inoperable turbine driven AFW pump in MODE 3 following a refueling is acceptable because the remaining motor driven AFW pumps remain capable of supplying additional redundant trains of AFW and the decay heat in the Reactor Coolant System is low. The probability of an event occurring during the extended outage time that would require the inoperable turbine driven AFW pump to function is low. The ACTION provides adequate assurance that the AFW System will continue to meet the assumptions stated in the safety analyses for the AFW system to mitigate postulated accidents. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.7.1.2 Action b states that with one AFW pump inoperable in MODE 1, 2, or 3, restore the inoperable AFW train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; or, be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours. CTS 3.7.1.2.a allows 7 days for restoration of an inoperable AFW pump steam supply which also results in inoperability of the AFW pump for which CTS 3.7.1.2.b requires restoration to OPERABLE status within 72 hours. ITS 3.7.5 ACTION B requires the restoration of one inoperable AFW train in MODES 1, 2, or 3 in 72 hours when the AFW train is inoperable for reasons other than Condition A. ITS LCO 3.7.5 ACTION A

states that with the turbine driven AFW train inoperable because of one inoperable steam supply or, if MODE 2 has not been entered following refueling, turbine driven AFW pump inoperable in MODE 3 following refueling, to restore affected equipment to OPERABLE status within 7 days. This changes the CTS by allowing a longer Completion Time (reasons other than Condition A) if the inoperable AFW train is the turbine driven AFW train or if the plant condition is following the refueling prior to entering MODE 2. (See DOC L01 for additional discussion)

The purpose of CTS 3.7.1.2, AFW system, is to provide redundant, independent, and diverse means of supplying feedwater to the SGs for cooling the RCS under emergency conditions. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features, the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. An inoperable turbine driven AFW pump in MODE 3 following a refueling is acceptable because the remaining motor driven AFW trains remain capable of supplying additional redundant trains of AFW and the decay heat in the Reactor Coolant System is low. The probability of an event occurring during the extended outage time that would require the turbine driven AFW pump to function is low. The ACTION provides adequate assurance that the AFW System will continue to meet the assumptions stated in the safety analyses for the AFW system to mitigate postulated accidents. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)*
CTS 4.7.1.2.b.1 requires that each automatic valve in the flow path actuates to its correct position upon receipt of actuation signal and CTS 4.7.1.2.b.2 requires that each auxiliary feedwater pump starts automatically as designed. In addition, Unit 1 CTS ITS 4.7.1.2.c and Unit 2 CTS 4.7.1.2.d require verification of AFW pump developed head. SR 3.7.5.3, SR 3.7.5.4, and SR 3.7.5.2 respectively, require similar Surveillances. However, they are modified by a Note that states the SR is not required to be performed for the turbine driven AFW pump until 24 hours after entering MODE 3. This changes the CTS by removing the specific text “and prior to entering MODE 2” from the surveillance requirement and providing an allowance to delay the performance of required testing without requiring the associated AFW train to be declared inoperable. R1

The purpose of CTS 4.7.1.2.b.1, CTS 4.7.1.2.b.2, CTS 4.7.1.2.c (Unit 1), and CTS 4.7.1.2.d (Unit 2) is to ensure the AFW train is OPERABLE in MODES 1, 2, and 3. The allowance provides for entry into MODE 3 before requiring the testing of the turbine driven AFW pump automatic valves or automatic pump start. This change is acceptable because the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. This change is necessary because the steam generator pressure may be insufficient in MODE 4 to properly test the R1

AFW train. One manner of performing this SR would be to combine the tests. The majority of SRs demonstrate equipment is, in fact, OPERABLE when the tests are performed. Inconsistent testing results may occur if testing of the AFW train is required before establishing sufficient conditions for the test. The allowance will permit the establishment of stable unit conditions and sufficient steam generator pressure to test the turbine driven AFW pump and valves and will allow an accurate and consistent method for the testing. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L04 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.7.1.2.b.1 requires the verification that each automatic valve in the flow path actuates to its correct position. ITS SR 3.7.5.3 requires verifying that each AFW automatic valve "not locked, sealed, or otherwise secured in position," actuates to the correct position. This changes the CTS by only requiring the testing of AFW valves that are not locked, sealed or otherwise secured in position.

The purpose of CTS 4.7.1.2.b.1 is to verify that the automatic valves in the AFW System flow paths align to the correct position. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. The testing of automatic valves that are aligned and secured into the required safety position is unnecessary. Valves secured in the safety position will satisfy the safety analyses assumptions for the mitigation of analyzed accidents. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L05 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.7.1.2.b.1 and CTS 4.7.1.2.b.2 require verification of actuation of automatic auxiliary feedwater components automatically upon receipt of an Auto Start actuation “test” signal. ITS SR 3.7.5.3 and SR 3.7.5.4 specify that the signal may be from either an "actual or simulated actuation” signal.

The purpose of CTS 4.7.1.2.b.1 and CTS 4.7.1.2.b.2 is to ensure that the auxiliary feedwater components operate correctly upon receipt of an actuation signal. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual," "simulated," or "test" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

3.7.1.2 LCO 3.7.5 [Three] AFW trains shall be OPERABLE. 1

----------NOTE----------
DOC M01 Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

Applicability DOC M01 APPLICABILITY: MODES 1, 2, and 3,
[MODE 4 when steam generator is relied upon for heat removal]. 1

ACTIONS

----------NOTE----------
Action f LCO 3.0.4.b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a A. [Turbine driven AFW train inoperable due to one inoperable steam supply. OR ----------NOTE---------- Only applicable if MODE 2 has not been entered following refueling. ---------- DOC L01 T ~~One t~~urbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days [OR In accordance with the Risk Informed Completion Time Program]] 1 2

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b DOC L02 B. One AFW train inoperable in MODE 1, 2, or 3 [for reasons other than Condition A].	B.1 Restore AFW train to OPERABLE status.	72 hours [OR In accordance with the Risk Informed Completion Time Program] (1)
Action c C. Turbine driven AFW train inoperable due to one inoperable steam supply. AND One motor driven AFW train inoperable.	C.1 Restore the steam supply to the turbine driven train to OPERABLE status. OR C.2 Restore the motor driven AFW train to OPERABLE status.	[24 or 48] hours [24 or 48] hours (1)
Action a Action b Action c D. Required Action and associated Completion Time of Condition A [, B, or C] not met. [OR Action d [Two] AFW trains inoperable in MODE 1, 2, or 3 for reasons other than Condition C.]	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours (12) [18] hours (1)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action e and NOTE E. ~~[~~ ~~[~~Three~~]~~ AFW trains inoperable in MODE 1, 2, or 3.	E.1 ----------NOTE---------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------- Initiate action to restore one AFW train to OPERABLE status.	Immediately~~]~~ (1)
DOC M01 F. Required AFW train inoperable in MODE 4.	F.1 ----------NOTE---------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------- Initiate action to restore one AFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.7.1.2.a.1 SR 3.7.5.1	Verify each AFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (1)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until ~~[~~24~~]~~ hours after ~~reaching [800] psig in the steam generators~~. entering MODE 3 ----------	1 2
4.7.1.2.c DOC M01	Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.5.3	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until ~~[~~24~~]~~ hours after ~~reaching [800] psig in the steam generators~~. entering MODE 3 2. Not required to be met in MODE 4 when steam generator is relied upon for heat removal. ----------	2
DOC L03 DOC M01		
4.7.1.2.b.1 DOC L04	Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 1

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.4	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until ~~[~~24~~]~~ hours after ~~reaching [800] psig in the steam generators~~. entering MODE 3 2. Not required to be met in MODE 4 when steam generator is relied upon for heat removal. ----------	
4.7.1.2.b.2 DOC L05	Verify each AFW pump starts automatically on an actual or simulated actuation signal when in MODE 1, 2, or 3.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
SR 3.7.5.5	Verify the proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days

DOC L03
DOC M01
DOC M02
2
1

CTS

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

3.7.1.2 LCO 3.7.5 [Three] AFW trains shall be OPERABLE. 1

----------NOTE----------
DOC M01 Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

Applicability DOC M01 APPLICABILITY: MODES 1, 2, and 3,
[MODE 4 when steam generator is relied upon for heat removal]. 1

ACTIONS

----------NOTE----------
Action f LCO 3.0.4.b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a A. [Turbine driven AFW train inoperable due to one inoperable steam supply. OR ----------NOTE---------- Only applicable if MODE 2 has not been entered following refueling. ---------- DOC L01 T One turbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days [OR In accordance with the Risk Informed Completion Time Program]] 1 2

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b DOC L02 B. One AFW train inoperable in MODE 1, 2, or 3 [for reasons other than Condition A].	B.1 Restore AFW train to OPERABLE status.	72 hours [OR In accordance with the Risk Informed Completion Time Program] (1)
Action c C. Turbine driven AFW train inoperable due to one inoperable steam supply. AND One motor driven AFW train inoperable.	C.1 Restore the steam supply to the turbine driven train to OPERABLE status. OR C.2 Restore the motor driven AFW train to OPERABLE status.	[24 or 48] hours [24 or 48] hours (1)
Action a Action b Action c D. Required Action and associated Completion Time of Condition A [, B, or C] not met. [OR Action d [Two] AFW trains inoperable in MODE 1, 2, or 3 for reasons other than Condition C.]	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours (12) [18] hours (1)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action e and NOTE E. ~~[~~ ~~[~~Three~~]~~ AFW trains inoperable in MODE 1, 2, or 3.	E.1 ----------NOTE---------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------- Initiate action to restore one AFW train to OPERABLE status.	Immediately~~]~~ (1)
DOC M01 F. Required AFW train inoperable in MODE 4.	F.1 ----------NOTE---------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------- Initiate action to restore one AFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.7.1.2.a.1 SR 3.7.5.1	Verify each AFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (1)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until ~~[~~24~~]~~ hours after ~~reaching [800] psig in the steam generators~~. entering MODE 3 ----------	1 2
4.7.1.2.c DOC M01	Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.5.3	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until ~~[~~24~~]~~ hours after ~~reaching [800] psig in the steam generators~~. entering MODE 3 2. Not required to be met in MODE 4 when steam generator is relied upon for heat removal. ----------	2
DOC L03 DOC M01		
4.7.1.2.b.1 DOC L04	Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 1

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.4	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until ~~[~~24~~]~~ hours after ~~reaching [800] psig in the steam generators~~. entering MODE 3 2. Not required to be met in MODE 4 when steam generator is relied upon for heat removal. ----------	
4.7.1.2.b.2 DOC L05	Verify each AFW pump starts automatically on an actual or simulated actuation signal when in MODE 1, 2, or 3.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
4.7.1.2.c SR 3.7.5.5	Verify the proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days

DOC L03

DOC M01

2

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.5, AUXILIARY FEEDWATER (AFW) SYSTEM

1. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction ~~through separate and independent suction lines~~ from the condensate storage tank (CST) (LCO 3.7.6, "Condensate Storage Tank (CST)") and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or atmospheric dump valves (ADVs) (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). If the main condenser is available, steam may be released via the steam bypass valves and recirculated to the CST.

1

1

A normally closed line also allows Unit 1 AFW pumps to take suction from the Unit 2 CST if the Unit 1 CST is not available.

greater than full flow capacity

full capacity

A train consisting of a pump and its associated flow path.

The AFW System consists of [two] motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 100% of AFW flow capacity; the turbine driven pump provides 100% of the required capacity to the steam generators as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system.

2

1

the required

normally

Each motor driven AFW pump is powered from an independent Class 1E power supply, and feeds one steam generator, although each pump has the capability to be realigned from the control room to feed the other steam generator.

1

A cross connection is provided to enable the routing of flow from the motor-driven pumps to either steam generator. However, the full-flow capacity of the motor-driven AFW pumps does not assure delivery of 100% of the flow credited in the feedwater line break and main steam line break analyses by means of the cross-connection (accident analyses does not explicitly credit AFW flow via the cross-connection).

One pump at full flow is sufficient to remove decay heat and cool the unit to Shutdown Cooling (SDC) System entry conditions.

and relies solely on DC power for operation

The steam turbine driven AFW pump receives steam from either main steam header upstream of the main steam isolation valve (MSIV). Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump. The turbine driven AFW pump supplies a common header capable of feeding both steam generators, with DC powered control valves actuated to the appropriate steam generator by the ~~Emergency~~ Feedwater Actuation System (~~E~~FAS).

1

Auxiliary

A

The AFW System supplies feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

1

BASES

BACKGROUND (continued)

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to SDC entry conditions, and steam is released through the ADVs.

The AFW System actuates automatically on low steam generator level by the ~~E~~FAS as described in LCO 3.3.~~4~~, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." The ~~E~~FAS logic is designed to feed either or both steam generators with low levels, but will isolate the AFW System from a steam generator having a significantly lower steam pressure than the other steam generator. The ~~E~~FAS automatically actuates the AFW turbine driven pump and associated DC operated valves and controls when required, to ensure an adequate feedwater supply to the steam generators. ~~D~~DC operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System is discussed in the FSAR, Section ~~[~~10.~~4. 9]~~ (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat, by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest MSSV set pressure ~~plus 3%~~.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

a. Feedwater Line Break (FWLB) and

b. Loss of normal feedwater.

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident.

The AFW System design is such that it can perform its function following a~~n FWLB between the MFW isolation valve and containment,~~ combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. ~~In such a case, the EFAS logic might not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly~~. ~~One motor driven AFW pump would deliver to the broken MFW header at the pump runout~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~flow until the problem was detected, and flow was terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump~~.

In such a case, the motor driven pumps deliver sufficient flow to the steam generators.

The AFW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that ~~[~~three~~]~~ AFW trains be OPERABLE to ensure that the AFW System will perform the design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps, in two diverse trains, ensure availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two pumps from independent emergency buses. The third AFW pump is powered by a diverse means, a steam driven turbine supplied with steam from a source not isolated by the closure of the MSIVs.

The AFW System is considered to be OPERABLE when the components and flow paths required to provide AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to a separate steam generator. The turbine driven AFW pump shall be OPERABLE with redundant steam supplies from each of the two main steam lines upstream of the MSIVs and capable of supplying AFW flow to either of the two steam generators. The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE.

The LCO is modified by a Note indicating that only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of reduced heat removal requirements, the short period of time in MODE 4 during which AFW is required, and the insufficient steam supply available in MODE 4 to power the turbine driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE and to function in the event that ~~the MFW~~ normal feedwater is lost. In addition, the AFW System is required to supply enough makeup water to replace steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4, the AFW System may be used for heat removal via the steam generator.

In MODES 5 and 6, the steam generators are not normally used for decay heat removal, and the AFW System is not required.

BASES

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

~~[~~ A.1

If the turbine driven AFW pump is inoperable due to one inoperable steam supply, or if a turbine driven pump is inoperable for any reason while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The 7 day Completion Time is reasonable based on the following reasons: 2

a. For the inoperability of the turbine driven AFW pump due to one inoperable steam supply, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump and the turbine driven train is still capable of performing its specified function for most postulated events.

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling outage, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

c. For both the inoperability of the turbine driven pump due to one inoperable steam supply and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps; and due to the low probability of an event requiring the use of the turbine driven AFW pump.

Condition A is modified by a Note which limits the applicability of the Condition for an inoperable turbine driven AFW pump in MODE 3 to when the unit has not entered MODE 2 following a refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.~~]~~ 2

BASES

ACTIONS (continued)

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 [for reasons other than Condition A], action must be taken to restore OPERABLE status within 72 hours [or in accordance with the Risk Informed Completion Time Program]. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and flow paths remain to supply feedwater to the steam generators. 2

C.1 and C.2

With one of the required motor driven AFW trains (pump or flow path) inoperable and the turbine driven AFW train inoperable due to one inoperable steam supply, action must be taken to restore the affected equipment to OPERABLE status within [24] [48] hours. Assuming no single active failures when in this condition, the accident (a FLB or MSLB) could result in the loss of the remaining steam supply to the turbine driven AFW pump due to the faulted steam generator (SG). In this condition, the AFW system may no longer be able to meet the required flow to the SGs assumed in the safety analysis[, [either due to the analysis requiring flow from two AFW pumps or due to the remaining AFW pump having to feed a faulted SG]]. 2 2

---REVIEWER'S NOTE---

Licensees should adopt the appropriate Completion Time based on their plant design. The 24 hour Completion Time is applicable to plants that can no longer meet the safety analysis requirement of 100% AFW flow to the SG(s) assuming no single active failure and a FLB or MSLB resulting in the loss of the remaining steam supply to the turbine driven AFW pump. The 48 hour Completion Time is applicable to plants that can still meet the safety analysis requirement of 100% AFW flow to the SG(s) assuming no single active failure and a FLB or MSLB resulting in the loss of the remaining steam supply to the turbine driven AFW pump.

3

[The 24 hour Completion Time is reasonable based on the remaining OPERABLE steam supply to the turbine driven AFW pump, the availability of the remaining OPERABLE motor driven AFW pump, and the low probability of an event occurring that would require the inoperable steam supply to be available for the turbine driven AFW pump.]] 2

BASES

ACTIONS (continued)

~~[The 48 hour Completion Time is reasonable based on the fact that the remaining motor driven AFW train is capable of providing 100 % of the AFW flow requirements, and the low probability of an event occurring that would challenge the AFW system.]~~ 2

D.1 and D.2

When Required Action A.1, ~~[~~ B.1, C.1, or C.2~~]~~ cannot be completed within the required Completion Time, ~~[~~or if two AFW trains are inoperable in MODE 1, 2, or 3 for reasons other than Condition C~~]~~, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within ~~[18]~~ hours. 2 12

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4, with ~~[~~two AFW trains inoperable ~~in MODE 1, 2, or 3]~~, operation is allowed to continue because only one motor driven AFW pump is required in accordance with the Note that modifies the LCO. Although it is not required, the unit may continue to cool down and start the SDC. 2 System

E.1

Required Action E.1 is modified by a Note indicating that all required MODE changes are suspended until one AFW train is restored to OPERABLE status.

With all ~~[~~three~~]~~ AFW trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition. 2

BASES

ACTIONS (continued)

F.1

Required Action F.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status.

the required

With ~~one~~ AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status or to immediately verify, by administrative means, the OPERABILITY of a second train. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition.

In MODE 4, either the reactor coolant pumps or the SDC loops can be used to provide forced circulation as discussed in LCO 3.4.6, "RCS Loops - MODE 4."

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW water and steam supply flow paths provides assurance that the proper flow paths exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulations; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

~~[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

3

2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by the ASME Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, as discussed in the ASME Code (Ref. 2) and the INSERVICE TESTING PROGRAM, satisfies this requirement.

turbine driven AFW pump

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is an insufficient steam pressure to perform the test. 1

SR 3.7.5.2

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an EFAS signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. A 1

~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 3

~~]~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

turbine driven AFW pump

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions have been established. This deferral is required because there is an insufficient steam pressure to perform the test. 1

Also, this SR is modified by a Note that states the SR is not required to be met in MODE 4. In MODE 4, the required AFW train is already aligned and operating.

SR 3.7.5.4

A

This SR ensures that the AFW pumps will start in the event of any accident or transient that generates an ~~E~~FAS signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal~~. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 3

turbine driven AFW pump

~~[~~This SR is modified by two Notes. Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is acceptable

~~required~~ because there is insufficient steam pressure to perform the test. Note 2 states that the SR is not required to be met in MODE 4. ~~[~~In MODE 4, the required pump is already operating and the autostart function is not required and if a pump is not already operating

~~.] [In MODE 4,~~ the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.~~]~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~
~~Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.~~

3

SR 3.7.5.5

This SR ensures that the AFW System is properly aligned by verifying the flow path to each steam generator prior to entering MODE 2 operation, after 30 days in any combination of MODE 5 or 6, or defueled. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and other administrative controls to ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, the OPERABILITY of the flow paths is verified following extended outages to determine that no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned by requiring a verification of minimum flow capacity of ~~750~~ gpm at ~~1270 psi~~. ~~(This SR is not required by those units that use AFW for normal startup and shutdown.)~~

296

985 psia

1

REFERENCES

U

1. FSAR, Section ~~[~~10.~~4.9]~~.

5.2

1 2

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction ~~through separate and independent suction lines~~ from the condensate storage tank (CST) (LCO 3.7.6, "Condensate Storage Tank (CST)") and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or atmospheric dump valves (ADVs) (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). If the main condenser is available, steam may be released via the steam bypass valves and recirculated to the CST.

1

1

greater than full flow capacity

full capacity

A train consisting of a pump and its associated flow path.

The AFW System consists of ~~[~~two~~]~~ motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 100% of AFW flow capacity; the turbine driven pump provides 100% of the required capacity to the steam generators as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system.

the required

2

1

normally

Each motor driven AFW pump is powered from an independent Class 1E power supply, and feeds one steam generator, although each pump has the capability to be realigned ~~from the control room~~ to feed the other steam generator.

1

A cross connection is provided to enable the routing of flow from the motor-driven pumps to either steam generator. However, the full-flow capacity of the motor-driven AFW pumps does not assure delivery of 100% of the flow credited in the feedwater line break and main steam line break analyses by means of the cross-connection (accident analyses does not explicitly credit AFW flow via the cross-connection).

One pump at full flow is sufficient to remove decay heat and cool the unit to Shutdown Cooling (SDC) System entry conditions.

and relies solely on DC power for operation

The steam turbine driven AFW pump receives steam from either main steam header upstream of the main steam isolation valve (MSIV). Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump. The turbine driven AFW pump supplies a common header capable of feeding both steam generators, with DC powered control valves actuated to the appropriate steam generator by the ~~Emergency~~ Feedwater Actuation System (~~E~~FAS).

1

Auxiliary

A

The AFW System supplies feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

1

BASES

BACKGROUND (continued)

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to SDC entry conditions, and steam is released through the ADVs.

The AFW System actuates automatically on low steam generator level by the ~~E~~EFAS as described in LCO 3.3.~~4~~, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." The ~~E~~EFAS logic is designed to feed either or both steam generators with low levels, but will isolate the AFW System from a steam generator having a significantly lower steam pressure than the other steam generator. The ~~E~~EFAS automatically actuates the AFW turbine driven pump and associated DC operated valves and controls when required, to ensure an adequate feedwater supply to the steam generators. ~~DC~~ operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System is discussed in the FSAR, Section ~~[~~10.4. 9~~]~~ (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat, by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest MSSV set pressure ~~plus 3%~~.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

a. Feedwater Line Break (FWLB) and

b. Loss of normal feedwater.

In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident.

The loss of main feedwater is the limiting event in terms of decay heat removal by the AFW system.

loss of main feedwater and FWLB events both with and without offsite power

The AFW System design is such that it can perform its function following an ~~FWLB between the MFW isolation valve and containment, combined with a loss of offsite power following turbine trip,~~ and a single active failure of the steam turbine driven AFW pump. ~~In such a case, the EFAS logic might not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly~~. ~~One motor driven AFW pump would deliver to the broken MFW header at the pump runout~~

The FWLB event with a non-isolable break between the steam generator and feedwater line check valve results in the lowest capacity of AFW to the intact steam generator

BASES

APPLICABLE SAFETY ANALYSES (continued)

In such a case, a faulted steam generator is sensed by steam generator pressure differences and main feedwater pressure differences and the available motor driven pump delivers sufficient flow to the intact steam generator.

~~flow until the problem was detected, and flow was terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump~~. 1

The AFW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that ~~[~~three~~]~~ AFW trains be OPERABLE to ensure that the AFW System will perform the design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps, in two diverse trains, ensure availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two pumps from independent emergency buses. The third AFW pump is powered by a diverse means, a steam driven turbine supplied with steam from a source not isolated by the closure of the MSIVs. 2

The AFW System is considered to be OPERABLE when the components and flow paths required to provide AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to a separate steam generator. The turbine driven AFW pump shall be OPERABLE with redundant steam supplies from each of the two main steam lines upstream of the MSIVs and capable of supplying AFW flow to either of the two steam generators. The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE.

The LCO is modified by a Note indicating that only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of reduced heat removal requirements, the short period of time in MODE 4 during which AFW is required, and the insufficient steam supply available in MODE 4 to power the turbine driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE and to function in the event that ~~the~~ ~~MFW~~ normal feedwater is lost. In addition, the AFW System is required to supply enough makeup water to replace steam generator secondary inventory, lost as the unit cools to MODE 4 conditions. 1

In MODE 4, the AFW System may be used for heat removal via the steam generator.

In MODES 5 and 6, the steam generators are not normally used for decay heat removal, and the AFW System is not required.

BASES

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

~~[~~ A.1

If the turbine driven AFW pump is inoperable due to one inoperable steam supply, or if a turbine driven pump is inoperable for any reason while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. The 7 day Completion Time is reasonable based on the following reasons: 2

a. For the inoperability of the turbine driven AFW pump due to one inoperable steam supply, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump and the turbine driven train is still capable of performing its specified function for most postulated events.

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling outage, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

c. For both the inoperability of the turbine driven pump due to one inoperable steam supply and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps; and due to the low probability of an event requiring the use of the turbine driven AFW pump.

Condition A is modified by a Note which limits the applicability of the Condition for an inoperable turbine driven AFW pump in MODE 3 to when the unit has not entered MODE 2 following a refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.~~]~~ 2

BASES

ACTIONS (continued)

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 [for reasons other than Condition A], action must be taken to restore OPERABLE status within 72 hours [or in accordance with the Risk Informed Completion Time Program]. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and flow paths remain to supply feedwater to the steam generators. 2

C.1 and C.2

With one of the required motor driven AFW trains (pump or flow path) inoperable and the turbine driven AFW train inoperable due to one inoperable steam supply, action must be taken to restore the affected equipment to OPERABLE status within [24] [48] hours. Assuming no single active failures when in this condition, the accident (a FLB or MSLB) could result in the loss of the remaining steam supply to the turbine driven AFW pump due to the faulted steam generator (SG). In this condition, the AFW system may no longer be able to meet the required flow to the SGs assumed in the safety analysis, [either due to the analysis requiring flow from two AFW pumps or due to the remaining AFW pump having to feed a faulted SG]. 2 2

---------- REVIEWER'S NOTE ----------
Licensees should adopt the appropriate Completion Time based on their plant design. The 24 hour Completion Time is applicable to plants that can no longer meet the safety analysis requirement of 100% AFW flow to the SG(s) assuming no single active failure and a FLB or MSLB resulting in the loss of the remaining steam supply to the turbine driven AFW pump. The 48 hour Completion Time is applicable to plants that can still meet the safety analysis requirement of 100% AFW flow to the SG(s) assuming no single active failure and a FLB or MSLB resulting in the loss of the remaining steam supply to the turbine driven AFW pump.
---------- 3

[The 24 hour Completion Time is reasonable based on the remaining OPERABLE steam supply to the turbine driven AFW pump, the availability of the remaining OPERABLE motor driven AFW pump, and the low probability of an event occurring that would require the inoperable steam supply to be available for the turbine driven AFW pump.]] 2

BASES

ACTIONS (continued)

~~[The 48 hour Completion Time is reasonable based on the fact that the remaining motor driven AFW train is capable of providing 100 % of the AFW flow requirements, and the low probability of an event occurring that would challenge the AFW system.]~~ 2

D.1 and D.2

When Required Action A.1, ~~[~~ B.1, C.1, or C.2~~]~~ cannot be completed within the required Completion Time, ~~[~~or if two AFW trains are inoperable in MODE 1, 2, or 3 for reasons other than Condition C~~]~~, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within ~~[18]~~ hours. 2 12

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4, with ~~[~~two AFW trains inoperable ~~in MODE 1, 2, or 3]~~, operation is allowed to continue because only one motor driven AFW pump is required in accordance with the Note that modifies the LCO. Although it is not required, the unit may continue to cool down and start the SDC. 2 System

E.1

Required Action E.1 is modified by a Note indicating that all required MODE changes are suspended until one AFW train is restored to OPERABLE status.

With all ~~[~~three~~]~~ AFW trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition. 2

BASES

ACTIONS (continued)

F.1

Required Action F.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status.

the required

With ~~one~~ AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status or to immediately verify, by administrative means, the OPERABILITY of a second train. LCO 3.0.3 is not applicable, as it could force the unit into a less safe condition.

In MODE 4, either the reactor coolant pumps or the SDC loops can be used to provide forced circulation as discussed in LCO 3.4.6, "RCS Loops - MODE 4."

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW water and steam supply flow paths provides assurance that the proper flow paths exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulations; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

~~[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

3

2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by the ASME Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, as discussed in the ASME Code (Ref. 2) and the INSERVICE TESTING PROGRAM, satisfies this requirement.

turbine driven AFW pump

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is an insufficient steam pressure to perform the test. 1

SR 3.7.5.3

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an ~~E~~EFAS signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. A 1

~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 3

~~]~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

turbine driven AFW pump

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions have been established. This deferral is required because there is an insufficient steam pressure to perform the test.

Also, this SR is modified by a Note that states the SR is not required to be met in MODE 4. In MODE 4, the required AFW train is already aligned and operating.

SR 3.7.5.4

A

This SR ensures that the AFW pumps will start in the event of any accident or transient that generates an ~~E~~FAS signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal~~. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

turbine driven AFW pump

~~[~~This SR is modified by two Notes. Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is ~~required~~ because there is insufficient steam pressure to perform the test. Note 2 states that the SR is not required to be met in MODE 4. ~~[~~In MODE 4, the required pump is already operating and the autostart function is not required~~.] [In MODE 4,~~ the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.~~]~~

acceptable

and if a pump is not already operating

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.~~

3

SR 3.7.5.5

This SR ensures that the AFW System is properly aligned by verifying the flow path to each steam generator prior to entering MODE 2 operation, after 30 days in any combination of MODE 5 or 6, or defueled. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and other administrative controls to ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, the OPERABILITY of the flow paths is verified following extended outages to determine that no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned by requiring a verification of minimum flow capacity of 275 ~~750~~ gpm at 1000 psia ~~1270 psi~~. ~~(This SR is not required by those units that use AFW for normal startup and shutdown.)~~

1

REFERENCES

U

1. FSAR, Section ~~[~~10.4.9~~]~~.

1 2

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.5 BASES, AUXILIARY FEEDWATER (AFW) SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.5, AUXILIARY FEEDWATER (AFW) SYSTEM

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

ITS 3.7.6, Condensate Storage Tank (CST)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

LCO 3.7.6
SR 3.7.6.1

3.7.1.3 The condensate storage tank shall be OPERABLE with a minimum contained volume of 153,400 gallons.

Applicability

APPLICABILITY: MODES 1, 2 and 3~~.~~ ,

Add proposed second Applicability

M01

ACTION:

Add proposed Required Action A.1

L01

ACTION A

With the condensate storage tank inoperable, restore the condensate storage tank to OPERABLE status within ~~4 hours~~ or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

7 days

ACTION B

M01

MODE 3

12 hours

MODE 4 without reliance on steam generator for heat removal

SURVEILLANCE REQUIREMENTS

Verify CST level ≥ 153,400 gallons

SR 3.7.6.1

4.7.1.3 ~~The condensate storage tank shall be demonstrated OPERABLE~~ in accordance with the Surveillance Frequency Control Program by verifying the water level.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

LCO 3.7.6
SR 3.7.6.1

3.7.1.3 The condensate storage tank (CST #2) shall be OPERABLE with a contained volume of at least 307,000 gallons.

Applicability

APPLICABILITY: MODES 1, 2, and 3~~.~~ , Add proposed second Applicability

M01

ACTION:

Add proposed Required Action A.1 and

ACTION A

With the condensate storage tank inoperable, within 4 hours restore the CST to OPERABLE status or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and in in 7 days

L01

ACTION B

~~HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours. MODE 3

MODE 4 without reliance on steam generator for heat removal

12

M01

SURVEILLANCE REQUIREMENTS

Verify CST level ≥ 307,000 gallons

SR 3.7.6.1

4.7.1.3 ~~The condensate storage tank shall be demonstrated OPERABLE~~ in accordance with the Surveillance Frequency Control Program ~~by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps~~.

A02

DISCUSSION OF CHANGES
ITS 3.7.6, CONDENSATE STORAGE TANK (CST)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 2 Only**: CTS 4.7.1.3 states, in part, that the CST shall be demonstrated OPERABLE when the tank is the supply source for the auxiliary feedwater pumps. ITS SR 3.7.6.1 states that the CST level must be verified to be within the specified limit. This changes the CTS by not stating that the Surveillance must be performed when the CST is the supply source for the auxiliary feedwater pumps.

The purpose of CTS 4.7.1.3 is to ensure the CST is OPERABLE when it is the supply source for the auxiliary feedwater pumps. Surveillance requirements are not required to be performed on inoperable equipment. ITS SR 3.0.1 states "Surveillances do not have to be performed on inoperable equipment or variables outside specified limits." If the CST is not capable of supplying the auxiliary feedwater pumps, the CST is inoperable because it can no longer provide its required support function. Furthermore, since inoperable equipment does not have to be tested, the removal of the phrase "when the tank is the supply source for the auxiliary feedwater pumps" is acceptable. This change is designated as administrative because it does not result in technical changes to the CTS and includes revising the action end state consistent with the Applicability change.

MORE RESTRICTIVE CHANGES

M01 CTS 3.7.1.3 requirements for CST OPERABILITY are applicable in MODES 1, 2, and 3. ITS 3.7.6 is applicable in MODES 1. 2, and 3, and MODE 4 when a steam generator (SG) is relied upon for heat removal. In addition, consistent with this change in Applicability, the requirement to be in “COLD SHUTDOWN within the following 30 hours” is changed to “MODE 4 without reliance on steam generator for heat removal within 12 hours” in ITS 3.7.6 Required Action B.2. This changes the CTS requirements by requiring the CST to be OPERABLE in MODE 4 when a steam generator is relied upon for heat removal and includes revising the action end state consistent with the Applicability change.

These changes are necessary because the CST may be needed in MODE 4 if shutdown cooling (SDC) has not been placed in service. If offsite power is lost when SDC is not yet in service, the SGs are supplied by the auxiliary feedwater pumps for heat removal. Therefore, the CST must be available in MODE 4 to support the Auxiliary Feedwater System. This change is designated as more restrictive because the CST is now required to be OPERABLE in MODE 4 when a SG is relied upon for heat removal.

DISCUSSION OF CHANGES
ITS 3.7.6, CONDENSATE STORAGE TANK (CST)

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.7.1.3 ACTION allows four hours to restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ITS 3.7.6 Required Action A.1 requires the verification of an OPERABLE backup water supply within 4 hours and once per 12 hours thereafter and Required Action A.2 requires the CST to be restored to OPERABLE status within 7 days. This changes the CTS by extending the Completion Time in the CTS 3.7.1.3 ACTION to restore the CST to OPERABLE status from 4 hours to 7 days provided the backup water supply is verified OPERABLE within 4 hours and every 12 hours thereafter.

The purpose of CTS 3.7.1.3 ACTION is to provide compensatory actions for when the CST is found to be inoperable. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to degraded conditions to minimize risk associated with continued operation while providing time to repair inoperable features and the low probability of an event requiring the use of the water from the CST occurring during this period. The backup supply to the Unit 1 CST is the Unit 2 CST and the backup supply to the Unit 2 CST is the primary water and city water storage tanks. The Completion Time to restore the CST to OPERABLE status considers that the backup supply may be performing this function in addition to its normal functions. Verification of a backup water supply with periodic re-verification provides adequate assurance that the CTS volume analysis assumptions are satisfied and continued operation is justified for the 7 day allowed restoration period. In addition, the opportunity to restore the equipment to OPERABLE status is always available. ITS LCO 3.0.2 states that upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated. Therefore, based on ITS LCO 3.0.2 restoration is always an option. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

3.7.1.3 LCO 3.7.6 The CST shall be OPERABLE.

Applicability DOC M01 APPLICABILITY: MODES 1, 2, and 3,
~~[~~MODE 4 when steam generator is relied upon for heat removal~~]~~. (2)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION DOC L01 A. CST inoperable.	A.1 Verify ~~OPERABILITY~~ of backup water supply. (by administrative means availability)	4 hours (3) | R1 AND Once per 12 hours thereafter
	AND A.2 Restore CST to OPERABLE status.	7 days
ACTION DOC M01 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 without reliance on steam generator for heat removal.	6 hours (12) ~~[24]~~ hours (2)

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.7.1.3 SR 3.7.6.1	Verify CST level is ≥ ~~[350,000]~~ 153,400 gal.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ (2)

CTS

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

3.7.1.3 LCO 3.7.6 The CST shall be OPERABLE.

Applicability DOC M01 APPLICABILITY: MODES 1, 2, and 3,
[MODE 4 when steam generator is relied upon for heat removal].

2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION DOC L01 A. CST inoperable.	A.1 Verify ~~OPERABILITY~~ of backup water supply. by administrative means availability	4 hours AND Once per 12 hours thereafter
	AND	
	A.2 Restore CST to OPERABLE status.	7 days
ACTION DOC M01 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND	6 hours
	B.2 Be in MODE 4 without reliance on steam generator for heat removal.	12 [~~24~~] hours

3 | R1

2

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
4.7.1.3	SR 3.7.6.1	Verify CST level is ≥ ~~[350,000]~~ 307,000 gal.	~~[~~ 12 hours ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.6, CONDENSATE STORAGE TANK (CST)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant information/value is inserted to reflect the current licensing basis.

3. Improved Standard Technical Specifications (ISTS) Required Action A.1 requires verification of the OPERABILITY of a backup water supply. As discussed in NEI 18-03, the industry's OPERABILITY guidance, "OPERABILITY" only applies to SSCs that a TS LCO requires to be OPERABLE. Therefore, the OPERABILITY has been changed to that availability of the backup water supply must be verified. R1

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST provides a ~~safety grade~~ qualified source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.4, "Auxiliary Feedwater (AFW) System"). The steam produced is released to the atmosphere by the main steam safety valves (MSSVs) or the atmospheric dump valves (ADVs). The AFW pumps operate with a continuous recirculation to the CST.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam bypass valves. The condensed steam ~~is~~ may be returned to the CST via the condensate ~~by transfer pump~~ System. ~~This has the advantage of conserving condensate while minimizing releases to the environment.~~

Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena. The CST is designed to Seismic Category I requirements to ensure availability of the feedwater supply. Feedwater is also available from ~~an alternate source~~ the Unit 2 CST and enclosed by a vertical concrete missile barrier.

A description of the CST is found in the U FSAR, Section ~~[~~9.2.~~6]~~ 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis, discussed in the U FSAR, Chapters ~~[~~6~~]~~ and ~~[~~15~~]~~ (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the analysis assumption is ~~generally [30] minutes~~ 1 hour at MODE 3, steaming through the ~~MSSVs~~ ADVs followed by a cooldown to shutdown cooling (SDC) entry conditions at the ~~design~~ cooldown rate of 75°F/hr.

The limiting event for the condensate volume is the large feedwater line break with a coincident loss of offsite power. Single failures that also affect this event include the following:

a. The failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine) and

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. The failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

~~A nonlimiting event considered in CST inventory determinations is a break either in the main feedwater, or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, as the Emergency Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated by the retaining of steam generator inventory.~~ 1

design requirements and

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

8 hours

100.3

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for ~~[30 minutes]~~ following a reactor trip from ~~102~~% RTP, and ~~then~~ cool down the RCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during the cooldown, as well as to account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line. 2

The CTS volume minimum of 153,400 gallons includes an unusable volume of 12,055 gallons and an additional margin of 10,845 gallons. In addition, the Unit 2 CTS is required to maintain a dedicated backup volume of ≥ 130,500 gallons available for Unit 1 use if the Unit 1 CST is made inoperable due to a vertical missile.

volume

equivalent to

≥ 130,500

1

The CST ~~level~~ required is a usable volume of ~~≤ [350,000]~~ gallons, which is based on holding the unit in MODE 3 for ~~[4]~~ hour~~s~~, followed by a cooldown to SDC entry conditions at 75°F per hour. This ~~basis is established by the NRC Standard Review Plan Branch Technical Position, Reactor Systems Branch 5-1 (Ref. 4) and~~ exceeds the volume required by the accident analysis. 2 1

OPERABILITY of the CST is determined by maintaining the tank ~~level~~ at or above the minimum required ~~level~~. 1

volume

APPLICABILITY

In MODES 1, 2, and 3, ~~[~~and in MODE 4, when steam generator is being relied upon for heat removal,~~]~~ the CST is required to be OPERABLE. 2

In MODES 5 and 6, the CST is not required because the AFW System is not required.

BASES

ACTIONS

A.1 and A.2

availability

If the CST is not OPERABLE, the ~~OPERABILITY~~ of the backup water supply must be verified by administrative means within 4 hours and once every 12 hours thereafter. 4 R1

Availability

~~OPERABILITY~~ of the backup ~~feedwater~~ supply must include verification of the ~~OPERABILITY of flow paths from the backup supply to the AFW pumps, and availability of the~~ required volume of water in the backup supply. The CST must be returned to OPERABLE status within 7 days, as the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on a~~n OPERABLE~~ backup water supply being available, and the low probability of an event requiring the use of the water from the CST occurring during this period. 4 R1 1

(Unit 2 CST volume ≥ 130,500 gallons)

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within ~~[24]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. 2

12

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. (This ~~level~~ ≥ ~~[350,000]~~ gallons.) ~~[The 12 hour Frequency is based on operating experience, and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal CST level deviations.~~ 1 2

volume

153,400

BASES

SURVEILLANCE REQUIEMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 3 2

REFERENCES

1. FSAR, Section ~~[~~9.2.~~6]~~. 8
2. FSAR, Chapter ~~[~~6~~]~~. U 2 1
3. FSAR, Chapter ~~[~~15~~]~~. U

~~4. NRC Standard Review Plan Branch Technical Position RSB 5-1.~~ 1

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

qualified

BACKGROUND

The CST provides a ~~safety grade~~ source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.4, "Auxiliary Feedwater (AFW) System"). The steam produced is released to the atmosphere by the main steam safety valves (MSSVs) or the atmospheric dump valves (ADVs). The AFW pumps operate with a continuous recirculation to the CST. 1

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam bypass valves. The condensed steam ~~is~~ may be returned to the CST via the condensate ~~transfer pump~~ System. ~~This has the advantage of conserving condensate while minimizing releases to the environment.~~ 1

Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena. The CST is designed to Seismic Category I requirements and enclosed by a vertical concrete missile barrier to ensure availability of the feedwater supply. Feedwater is also available from ~~an~~ alternate sources; primary water and city water storage tanks provide an alternate source of makeup water to the CST. 1

A description of the CST is found in the U FSAR, Section ~~[~~9.2.6~~]~~ (Ref. 1). 2

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis, discussed in the FSAR, Chapters ~~[~~6~~]~~ and ~~[~~15~~]~~ (Refs. 2 and 3, respectively). For U anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the analysis assumption is 4 hours ~~generally [30] minutes~~ at MODE 3, steaming through the ~~MSSVs~~ ADVs followed by a cooldown to shutdown cooling (SDC) entry conditions at the ~~design~~ cooldown rate of 75°F/hr and equates to 154,000 gallons. 2

The limiting event for the condensate volume is the loss of one DC emergency train ~~large feedwater line break~~ with a coincident loss of offsite power. Single failures that also affect this event include the following: 1

a. The failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine) and

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. The failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

~~A nonlimiting event considered in CST inventory determinations is a break either in the main feedwater, or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, as the Emergency Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated by the retaining of steam generator inventory.~~ 1

design requirements and

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

11.4 hours

100.3

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for ~~[30 minutes]~~ following a reactor trip from ~~102~~% RTP, and ~~then~~ cool down the RCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during the cooldown, as well as to account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line. 2 1

volume

equivalent to

≥ 154,000

The CST ~~level~~ required is a usable volume of ~~≤ [350,000]~~ gallons, which is based on holding the unit in MODE 3 for ~~[~~4~~]~~ hours, followed by a cooldown to SDC entry conditions at 75°F per hour. ~~This basis is established by the NRC Standard Review Plan Branch Technical Position, Reactor Systems Branch 5-1 (Ref. 4) and exceeds the volume required by the accident analysis.~~ 2 1

The CTS minimum volume of 307,000 gallons includes an unusable volume of 13,433 gallons, 154,000 gallons dedicated to Unit 2, 130,500 reserved for Unit 1 CST backup and an additional margin of 9,067 gallons.

OPERABILITY of the CST is determined by maintaining the tank ~~level~~ at or above the minimum required ~~level~~. 1

volume

APPLICABILITY

In MODES 1, 2, and 3, ~~[~~and in MODE 4, when steam generator is being relied upon for heat removal,~~]~~ the CST is required to be OPERABLE. 2

In MODES 5 and 6, the CST is not required because the AFW System is not required.

BASES

ACTIONS

A.1 and A.2

availability

If the CST is not OPERABLE, the ~~OPERABILITY~~ of the backup water supply must be verified by administrative means within 4 hours and once every 12 hours thereafter. 4 R1

Availability

~~OPERABILITY~~ of the backup ~~feedwater~~ supply must include verification of the ~~OPERABILITY of flow paths from the backup supply to the AFW pumps, and availability of the~~ required volume of water in the backup supply. The CST must be returned to OPERABLE status within 7 days, as the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on a~~n OPERABLE~~ backup water supply being available, and the low probability of an event requiring the use of the water from the CST occurring during this period. 4 R1 1

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within ~~[24]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. 12 2

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. (This ~~level~~ ≥ ~~[350,000]~~ gallons.) ~~[The 12 hour Frequency is based on operating experience, and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal CST level deviations.~~ volume 307,000 1 2

BASES

SURVEILLANCE REQUIEMENTS (continued)

~~OR~~ (2)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ (3) (2)

REFERENCES

1. FSAR, Section ~~[~~9.2.6~~]~~.
2. FSAR, Chapter ~~[~~6~~]~~. (U) (2) (1)
3. FSAR, Chapter ~~[~~15~~]~~. (U)

~~4. NRC Standard Review Plan Branch Technical Position RSB 5-1.~~ (1)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.6 BASES, CONDENSATE STORAGE TANK (CST)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

4. Changes are made to be consistent with changes made to the Specification. | R1

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.6, CONDENSATE STORAGE TANK (CST)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

3.7.7, Component Cooling Water (CCW) System

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.7

~~3.7.3.1~~ ~~At least~~ two ~~independent~~ CCW ~~component cooling water~~ loops shall be OPERABLE. LA01

Applicability

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

ACTION A.1 Note

NOTE 3.4.6, "RCS Loops – MODE 4,"

Enter applicable ACTIONS of LCO ~~3.4.1.3, "Reactor Coolant System - Hot Shutdown,~~" for shutdown cooling loops made inoperable by CCW.

ACTION A

ACTION A.1

~~With only~~ one ~~component cooling water~~ CCW loop in OPERABLE, restore ~~at least two~~ CCW loop~~s~~ to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following~~ 6 hours. LCO 3.0.4.a is not applicable when entering ~~HOT SHUTDOWN~~.

ACTION B.1, B.2

ACTION B.2 Note

12 MODE 3 MODE 4 MODE 4

SURVEILLANCE REQUIREMENTS

~~4.7.3.1~~ ~~At least~~ two ~~component cooling water~~ CCW loops shall be demonstrated OPERABLE:

Add SR 3.7.7.1 Note A02

SR 3.7.7.1

~~a.~~ In accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing that~~ each ~~valve (~~manual, power operated ~~or~~ automatic~~)~~ servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

and valve in the flow path in the flow path A03

SR 3.7.7.2

~~b.~~ In accordance with the Surveillance Frequency Control Program ~~during shutdown by~~ verify~~ing that~~ each automatic valve servicing safety related equipment actuates to ~~its~~ the correct position on ~~a Safety Injection~~ an actual or simulated Actuation Signal. LA02 L01

that is not locked, sealed or otherwise secured in position L02

Add SR 3.7.7.3 M01

A01

~~**PLANT SYSTEMS**~~

~~**SURVEILLANCE REQUIREMENTS** (Continued)~~

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A01

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

~~NOTE~~

- ~~When CCW pump 2C is being used to satisfy the requirements of this specification, the alignment of the discharge valves shall be verified to be consistent with the appropriate power supply at least once per 24 hours.~~
- ~~Upon receipt of annunciation for improper alignment of the pump 2C motor power in relation to any of its motor-operated discharge valves positions, restore proper system alignment within 2 hours.~~
- Enter applicable ACTIONS of LCO ~~3.4.1.3, "Reactor Coolant System – Hot Shutdown,"~~ for shutdown cooling loops made inoperable by CCW.

L03

L04

ACTION A.1 Note

CCW

3.4.6, "RCS Loops – MODE 4,"

LCO 3.7.7 ~~3.7.3~~ ~~At least~~ two ~~independent~~ ~~component cooling water~~ loops shall be OPERABLE.

LA01

Applicability **APPLICABILITY:** MODES 1, 2, 3, and 4.

ACTION:

CCW

in

CCW

ACTION A ~~With only~~ one ~~component cooling water~~ loop OPERABLE, restore ~~at least two~~ loop~~s~~ to

ACTION A.1 OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within

ACTION B.1, B.2 ~~the following~~ ~~6~~ hours. LCO 3.0.4.a is not applicable when entering ~~HOT SHUTDOWN~~.

ACTION B.2 Note

12

MODE 3

MODE 4

MODE 4

SURVEILLANCE REQUIREMENTS

CCW

~~4.7.3~~ ~~At least~~ two ~~component cooling water~~ loops shall be demonstrated OPERABLE:

CCW

Add SR 3.7.7.1 Note

A02

SR 3.7.7.1 ~~a.~~ In accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing~~ ~~that~~ each ~~valve (~~manual, power-operated ~~or~~ automatic~~)~~ servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

and

valve in the flow path

in the flow path

A03

CCW

SR 3.7.7.2 ~~b.~~ In accordance with the Surveillance Frequency Control Program ~~during shutdown by~~ verify~~ing that~~ each automatic valve servicing safety-related equipment actuates to ~~its~~ correct position on ~~an SIAS test~~ signal.

LA02

the

an actual or simulated actuation

L01

that is not locked, sealed or otherwise secured in position

L02

Add SR 3.7.7.3

M01

DISCUSSION OF CHANGES
ITS 3.7.7, COMPONENT COOLING WATER (CCW) SYSTEM

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 4.7.3.1.a and Unit 2 4.7.3.a do not contain an explicit reference to isolating CCW flow to individual components. ITS SR 3.7.7.1 contains a Note which states, "Isolation of CCW flow to individual components does not render the CCW System inoperable." This changes CTS by adding an allowance that is not explicitly stated in the CTS. The purpose of the CCW System Technical Specification is to provide assurance that CCW is available to the appropriate plant components. This change is acceptable because by current use and application of the CTS, isolation of a component supplied with CCW does not necessarily result in the CCW System being considered inoperable, but the respective component may be declared inoperable for its system. This change clarifies this application. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 Unit 1 CTS 4.7.3.1.a and Unit 2 CTS 4.7.3.a require verification that each CCW valve (manual, power operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position. ITS SR 3.7.7.1 requires verification that each CCW manual, power operated, and automatic valve “in the flow path” servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in the correct position.

Unit 1 CTS 4.7.3.1.b and Unit 2 CTS 4.7.3.b require verification that each CCW automatic valve servicing safety related equipment actuates to its correct position on a safety injection actuation signal. ITS SR 3.7.7.2 requires verification that each CCW automatic valve “in the flow path” servicing safety related equipment that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actuation signal. This changes the CTS by adding the words "in the flow path" to Unit 1 CTS 4.7.3.1.a and Unit 2 CTS 4.7.3.a, and to Unit 1 CTS 4.7.3.1.b and Unit 2 CTS 4.7.3.b.

The purpose of Unit 1 CTS 4.7.3.1.a and Unit 2 CTS 4.7.3.a is to ensure all valves in the CCW System flow path are in the correct position. The purpose of Unit 1 CTS 4.7.3.1.b and Unit 2 CTS 4.7.3.b is to ensure all automatic valves in the CCW System flow path actuate to the correct position on an actual or simulated actuation signal. The addition of the words "in the flow path" does not change the intent of the Surveillance Requirements. Each manual, power operated, and automatic valve servicing safety related equipment that is not locked, sealed, or otherwise secured in position will continue to be verified to be in the correct position. Each CCW automatic valve in the flow path that is not

locked, sealed or otherwise secured in position, will still be checked to ensure it actuates to the correct position on an actual or simulated actuation signal. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 4.7.3.1 and Unit 2 CTS 4.7.3 do not contain a requirement to verify each CCW System pump starts automatically on an actuation signal. ITS SR 3.7.7.3 states; "Verify each CCW pump starts automatically on an actual or simulated actuation signal," with a Frequency specified in accordance with the Surveillance Frequency Control Program. This changes the CTS by adding a Surveillance Requirement to test the CCW System pumps.

This change is necessary because in order for the CCW System to perform the safety function assumed in the accident analysis, the CCW pumps must start automatically. The CCW System is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. Therefore, the SR periodically verifies proper automatic operation of the pumps on an actual or simulated actuation signal. This Surveillance is similar to the testing requirements for other safety related pumps. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, SR 3.7.7.3 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 18 months consistent with the ISTS SR 3.7.7.3 and is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This change is designated as more restrictive because it adds a new SR.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.7.3.1 and Unit 2 CTS 3.7.3 state that two "independent" CCW loops shall be OPERABLE. ITS 3.7.7 requires two CCW loops to be OPERABLE but does not contain the detail that the loops must be independent. This changes the CTS by moving the detail that the CCW loops are independent to the Bases. The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for two CCW loops to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification

Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 4.7.3.1.b and Unit 2 CTS 4.7.3.b require verification of the automatic actuation of CCW components on a safety injection actuation signal (SIAS). ITS SR 3.7.7.2 does not state the specific type of signal, but only specifies an actual or simulated "actuation" signal. This changes CTS by moving the type of actuation signal (i.e., SIAS) to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to verify that appropriate equipment actuates upon receipt of an actuation signal. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5 of the ITS. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Specification.

LESS RESTRICTIVE CHANGES

L01 *(Category 6- Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.7.3.1.b and Unit 2 CTS 4.7.3.b require verification of the automatic actuation of component cooling water components on a safety injection actuation signal (SIAS) "test" signal. ITS SR 3.7.7.2 specifies that the signal may be from either an "actual" or simulated (i.e., test) signal. This changes the CTS by explicitly allowing the use of either an actual or simulated signal for the test.

The purpose of Unit 1 CTS 4.7.3.1.b and Unit 2 CTS 4.7.3.b is to ensure the CCW components operate correctly upon receipt of an actuation signal. This change is acceptable because the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual," "simulated," or "test" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L02 *(Category 5 - Deletion of Surveillance Requirement)* Unit 1 CTS 4.7.3.1.b and Unit 2 CTS 4.7.3.b require verification that each CCW automatic valve in the flow path actuates to its correct position. ITS SR 3.7.7.2 requires verification that

each CCW automatic valve in the flow path "that is not locked, sealed, or otherwise secured in position" actuates to the correct position. This changes the CTS by excluding those valves that are locked, sealed, or otherwise secured in position from the verification.

The purpose of Unit 1 CTS 4.7.3.1.b and Unit 2 CTS 4.7.3.b is to provide assurance that if an event occurred requiring CCW valves to be in their correct position, then those requiring automatic actuation would actuate to their correct position. This change is acceptable because the deleted Surveillance is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested to in a manner and at a Frequency necessary to provide confidence that the equipment can perform its assumed safety function. Those automatic valves that are locked, sealed, or otherwise secured in position are not required to actuate on a safety injection actuation signal in order to perform their safety function because they are already in the required position. Testing such valves would not provide any additional assurance of OPERABILITY. Valves that are required to actuate will continue to be tested. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L03 **Unit 2 only:** *(Category 5 - Deletion of Surveillance Requirement)* CTS 3.7.3 note requires verification that when CCW pump 2C is being used to satisfy the requirements of this specification, the alignment of the discharge valves shall be verified to be consistent with the appropriate power supply at least once per 24 hours. ITS 3.7.7 does not require this specific verification. This changes the CTS by deleting the specific verification of valve and power supply alignment when the swing CCW pump is being used.

The purpose of the CTS 3.7.3 Note is to ensure that when the swing CCW pump (Pump 2C) is used to satisfy the requirements, the proper valve and power supply alignment is verified periodically. The PSL CCW System design includes three pumps: each pump provides flow to a redundant CCW loop. These two pumps are powered from the associated essential bus. The third swing pump can be powered from either essential bus and the discharge valves can be aligned to either CCW loop. When used to support CCW loop OPERABILITY, the swing pump power and valves must be aligned to the correct CCW loop. The Surveillances retained in ITS will continue to ensure the valves and breakers are properly aligned when the swing pump is being used. ITS SR 3.7.7.1 requires verification that the valves that are not locked sealed or otherwise secured in position (i.e., accident position) are in the correct position. This includes verifying the discharge valves of the swing pump are in their correct position to support CCW loop OPERABILITY. In addition, there is an alarm that annunciates in the control room in the event the discharge valves to the swing pump are not aligned to the same loop as the pump breaker is aligned. ITS SR 3.7.7.3 requires verification that the required pump automatically starts on an actual or simulated actuation signal. This test ensures the swing pump is powered from the correct essential bus by verifying the pump can automatically start on the selected CCW loop. The Frequency in accordance with the Surveillance Frequency Control Program (i.e., 31 days for the valve position verification and 18 months for the automatic pump start) has been determined a sufficient periodicity for verification

of required equipment. This change is acceptable because the deleted Surveillance Requirement is not necessary to ensure that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. This change is designated as less restrictive because a Surveillance which is required in the CTS will not be required in the ITS

L04 **Unit 2 only:** *(Category 3 – Relaxation of Completion Time)* CTS 3.7.3 note requires verification that upon receipt of annunciation for improper alignment of the pump 2C motor power in relation to any of its motor-operated discharge valves positions, restore proper system alignment within 2 hours. ITS SR 3.7.7 does not require this specific action. This changes the CTS by deleting the action to restore proper alignment of the swing CCW pump.

The purpose of the CTS 3.7.3 Note is to ensure that when the swing CCW pump (Pump 2C) is used to satisfy the requirements and the valve and power supply to the swing pump are not aligned, action is required to restore the alignment. The PSL CCW System design includes three pumps: two pumps each provide flow to a redundant CCW loop. These two pumps are powered from the associated essential bus. The third swing pump can be powered from either essential bus and the discharge valves can be aligned to either CCW loop. When used to support CCW loop OPERABILITY the swing pump power and valves must be aligned to the correct CCW loop. The ACTIONS retained in ITS will continue to ensure the appropriate remedial actions are taken when the swing pump is used to support CCW loop OPERABILITY. When a CCW loop is inoperable, ITS 3.7.7 ACTION A requires the CCW loop to be restored to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program. This action would include restoring the swing pump valve and power alignment when the swing pump is used to support OPERABILITY of the CCW loop. When the swing pump is not used to support OPERABILITY of a CCW loop, the requirement to restore alignment is not a required remedial action to restore the LCO as defined in 10 CFR 50.36(c)(2)(i). Therefore, the note is unnecessary. The ITS 3.7.7 ACTION A (CTS 3.7.3 Action) Completion Time has been determined sufficient to restore the CCW loop to OPERABLE status. This change is acceptable because the Completion Time of ITS 3.7.7 ACTION A is consistent with safe operation under the specified Condition, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

3.7.3.1 LCO 3.7.7 Two CCW ~~trains~~ loops shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.3.1 Action 3.7.3.1 Action Note A. One CCW ~~train~~ loop inoperable.	A.1 ----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for shutdown cooling loops made inoperable by CCW. ---------- Restore CCW ~~train~~ loop to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
3.7.3.1 Action B. Required Action and associated Completion Time ~~of Condition A~~ not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	----------NOTE---------- Isolation of CCW flow to individual components does not render the CCW System inoperable. ----------	
	Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
SR 3.7.7.2	Verify each CCW automatic valve in the flow path servicing safety related equipment that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
SR 3.7.7.3	Verify each required CCW pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

CTS

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

3.7.3.1 LCO 3.7.7 Two CCW ~~trains~~ loops shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.3 Action 3.7.3 Action Note A. One CCW ~~train~~ loop inoperable.	A.1 ----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for shutdown cooling loops made inoperable by CCW. ---------- Restore CCW ~~train~~ loop to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
3.7.3 Action B. Required Action and associated Completion Time ~~of Condition A~~ not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	----------NOTE---------- Isolation of CCW flow to individual components does not render the CCW System inoperable. ----------	
	Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
SR 3.7.7.2	Verify each CCW automatic valve in the flow path servicing safety related equipment that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
SR 3.7.7.3	Verify each required CCW pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.7, COMPONENT COOLING WATER (CCW) SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Since Condition B applies to all Conditions in the ACTIONS Table, the term "of Condition A" is not necessary. This is consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 4.1.6.i.5.ii.

4. The PSL design includes a spare CCW pump that can be substituted for one of the normal CCW pumps. Therefore SR 3.7.7.3 has been modified to only require the "required" CCW pumps to be tested. This is consistent with the use of the word required in the ITS, as discussed in the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 4.1.3.b.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

The surge tank is divided internally into two separate compartments. Each compartment serves one of the CCW loops.

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the ~~Service~~ Water System, and thus to the environment.

Intake Cooling

loop

The CCW System is arranged as two independent full capacity cooling loops, and has isolatable nonsafety related components. Each safety related ~~train~~ includes a full capacity pump, surge tank, heat exchanger, piping, valves, ~~and~~ instrumentation. Each safety related ~~train~~ is powered from a separate bus. An open surge tank in the system ~~provides pump trip protective functions~~ to ensure sufficient net positive suction head is available. The pump in each ~~train~~ is automatically started on receipt of a safety injection actuation signal, and all nonessential components are isolated.

, and a common

loop

is equipped with a low level alarm that annunciates in the control room

Three CCW pumps are provided. A pump can be removed from service without reducing the capability or redundancy of the CCW System. Two of the CCW pumps are powered from the associated essential bus. The third CCW pump can be powered from either essential bus.

loop

U

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section ~~[~~9.2.2~~]~~, Reference 1. The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Shutdown Cooling (SDC) System heat exchanger. This may utilize the ~~SCS~~ heat exchanger, during a normal or post accident cooldown and shutdown, or the Containment Spray System during the recirculation phase following a loss of coolant accident (LOCA).

SDC

loop

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one CCW ~~train~~ in conjunction with a 100% capacity Containment Cooling System (containment spray, containment coolers, or a combination) removing core decay heat 20 minutes after a design basis LOCA. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the safety injection pumps.

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

325

The CCW System also functions to cool the unit from SDC entry conditions (T_{cold} < ~~[~~350~~]~~°F) to MODE 5 (T_{cold} < ~~[~~200~~]~~°F) during normal and post accident operations. The time required to cool from ~~[350]~~°F to

325

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~[~~200~~]~~°F is a function of the number of CCW and SDC ~~trains~~ operating. One CCW ~~train~~ is sufficient to remove decay heat during subsequent operations with T_{cold} < ~~[~~200~~]~~°F. This assumes that a maximum seawater temperature of ~~76~~°F occurs simultaneously with the maximum heat loads on the system.

The CCW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The CCW ~~trains~~ are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CCW ~~train~~ is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two CCW ~~trains~~ must be OPERABLE. At least one CCW ~~train~~ will operate assuming the worst single active failure occurs coincident with the loss of offsite power.

A CCW ~~train~~ is considered OPERABLE when the following:

a. The associated pump and surge tank are OPERABLE and

b. The associated piping, valves, heat exchanger and instrumentation and controls required to perform the safety related function are OPERABLE.

The spare CCW pump can be substituted for a normal CCW pump, provided the spare CCW valves are aligned to the applicable loop and the power supply for the spare pump is aligned to the same essential bus as the pump it is replacing.

The isolation of CCW from other components or systems not required for safety may render ~~those~~ components or systems inoperable, but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system that must be prepared to perform its post accident safety functions, primarily RCS heat removal by cooling the SDC heat exchanger.

Although the LCO for the CCW System is not applicable in MODES 5 and 6, the capability of the CCW System to perform its necessary related support functions may be required for OPERABILITY of supported systems.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating the requirement of entry into the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for SDC made inoperable by CCW. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

BASES

ACTIONS (continued)

With one CCW ~~train~~ inoperable, action must be taken to restore OPERABLE status within 72 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. In this Condition, the remaining OPERABLE CCW ~~train~~ is adequate to perform the heat removal function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE ~~train~~, and the low probability of a DBA occurring during this period.

B.1 and B.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

If the CCW ~~train~~ cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is minimized. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. If CCW flow is lost to the reactor coolant pump (RCP) seals, entering MODE 5 and lowering the RCS temperature should be considered in order to avoid possible damage to the RCP seal materials.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable,

BASES

ACTIONS (continued)

because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

This SR is modified by a Note indicating that the isolation of the CCW components or systems may render those components inoperable but does not affect the OPERABILITY of the CCW System.

~~[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 2

1

U

REFERENCES

1. FSAR, Section ~~[~~9.2.2~~]~~.

2. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

2

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

The surge tank is divided internally into two separate compartments. Each compartment serves one of the CCW loops.

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the ~~Service~~ Water System, and thus to the environment.

Intake Cooling

loop

The CCW System is arranged as two independent full capacity cooling loops, and has isolatable nonsafety related components. Each safety related ~~train~~ includes a full capacity pump, surge tank, heat exchanger, piping, valves, ~~and~~ instrumentation. Each safety related ~~train~~ is powered from a separate bus. An open surge tank in the system ~~provides pump trip protective functions~~ to ensure sufficient net positive suction head is available. The pump in each ~~train~~ is automatically started on receipt of a safety injection actuation signal, and all nonessential components are isolated.

, and a common

loop

is equipped with a low level alarm that annunciates in the control room

Three CCW pumps are provided. A pump can be removed from service without reducing the capability or redundancy of the CCW System. Two of the CCW pumps are powered from the associated essential bus. The third CCW pump can be powered from either essential bus.

loop

U

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section ~~[~~9.2.2~~]~~, Reference 1. The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Shutdown Cooling (SDC) System heat exchanger. This may utilize the ~~SCS~~ heat exchanger, during a normal or post accident cooldown and shutdown, or the Containment Spray System during the recirculation phase following a loss of coolant accident (LOCA).

SDC

APPLICABLE SAFETY ANALYSES

loop

The design basis of the CCW System is for one CCW ~~train~~ in conjunction with a 100% capacity Containment Cooling System (containment spray, containment coolers, or a combination) removing core decay heat 20 minutes after a design basis LOCA. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the safety injection pumps.

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

325

The CCW System also functions to cool the unit from SDC entry conditions (T_{cold} < ~~[~~350~~]~~°F) to MODE 5 (T_{cold} < ~~[~~200~~]~~°F) during normal and post accident operations. The time required to cool from ~~[350]~~°F to

325

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~[~~200~~]~~°F is a function of the number of CCW and SDC ~~trains~~ operating. One CCW ~~train~~ is sufficient to remove decay heat during subsequent operations with T_{cold} < ~~[~~200~~]~~°F. This assumes that a maximum seawater temperature of ~~76~~°F occurs simultaneously with the maximum heat loads on the system.

The CCW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The CCW ~~trains~~ are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CCW ~~train~~ is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two CCW ~~trains~~ must be OPERABLE. At least one CCW ~~train~~ will operate assuming the worst single active failure occurs coincident with the loss of offsite power.

A CCW ~~train~~ is considered OPERABLE when the following:

a. The associated pump and surge tank are OPERABLE and

b. The associated piping, valves, heat exchanger and instrumentation and controls required to perform the safety related function are OPERABLE.

The spare CCW pump can be substituted for a normal CCW pump, provided the spare CCW valves are aligned to the applicable loop and the power supply for the spare pump is aligned to the same essential bus as the pump it is replacing.

The isolation of CCW from other components or systems not required for safety may render ~~those~~ components or systems inoperable, but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system that must be prepared to perform its post accident safety functions, primarily RCS heat removal by cooling the SDC heat exchanger.

Although the LCO for the CCW System is not applicable in MODES 5 and 6, the capability of the CCW System to perform its necessary related support functions may be required for OPERABILITY of supported systems.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating the requirement of entry into the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for SDC made inoperable by CCW. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

BASES

ACTIONS (continued)

With one CCW ~~train~~ inoperable, action must be taken to restore OPERABLE status within 72 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. In this Condition, the remaining OPERABLE CCW ~~train~~ is adequate to perform the heat removal function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE ~~train~~, and the low probability of a DBA occurring during this period.

B.1 and B.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

If the CCW ~~train~~ cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is minimized. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. If CCW flow is lost to the reactor coolant pump (RCP) seals, entering MODE 5 and lowering the RCS temperature should be considered in order to avoid possible damage to the RCP seal materials.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable,

BASES

ACTIONS (continued)

because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

This SR is modified by a Note indicating that the isolation of the CCW components or systems may render those components inoperable but does not affect the OPERABILITY of the CCW System.

~~[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

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REFERENCES

1. FSAR, Section ~~[~~9.2.2~~]~~.
2. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.7, BASES, COMPONENT COOLING WATER (CCW) SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.7, COMPONENT COOLING WATER (CCW) SYSTEM

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 8

3.7.8, Intake Cooling Water (ICW) System

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

PLANT SYSTEMS

(ICW)

3/4.7.4 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

(ICW)

LCO 3.7.8 ~~3.7.4.1~~ ~~At least~~ two ~~independent~~ ~~intake cooling water~~ loops shall be OPERABLE. LA01

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTION:

ACTION A.1 Note

> **NOTE**
>
> 3.4.6, "RCS Loops – MODE 4,"
>
> Enter applicable ACTIONS of LCO ~~3.4.1.3, "Reactor Coolant System - Hot Shutdown,"~~ for shutdown cooling loops made inoperable by ICW.

ICW in ICW

ACTION A ACTION A.1 ~~With only~~ one ~~intake cooling water~~ loop OPERABLE, restore ~~at least two~~ loop~~s~~ to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours. LCO 3.0.4.a is not applicable when entering ~~HOT SHUTDOWN~~.

ACTION B.1, B.2

ACTION B.2 Note

12 MODE 3 MODE 4 MODE 4

SURVEILLANCE REQUIREMENTS

ICW

~~4.7.4.1~~ ~~At least~~ two ~~intake cooling water~~ loops shall be demonstrated OPERABLE:

ICW Add SR 3.7.8.1 Note A02

SR 3.7.8.1 ~~a.~~ In accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing that~~ each ~~valve (~~manual, power operated ~~or~~ automatic~~)~~ servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

and valve in the flow path A03

ICW in the flow path

SR 3.7.8.2 ~~b.~~ In accordance with the Surveillance Frequency Control Program ~~during shutdown by~~ verify~~ing that~~ each automatic valve servicing safety related equipment actuates to ~~its~~ correct position on ~~a Safety Injection~~ Actuation signal. LA02

the an actual or simulated L01

that is not locked, sealed or otherwise secured in position L02

Add SR 3.7.8.3 M01

A01

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

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A01

PLANT SYSTEMS

(ICW)

3/4.7.4 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

NOTE

- ~~When ICW pump 2C is being used to satisfy the requirements of this specification, the alignment of the discharge valves must be verified to be consistent with the appropriate power supply at least once per 24 hours.~~ L03
- Enter applicable ACTIONS of ~~LCO 3.4.1.3, "Reactor Coolant System – Hot Shutdown,"~~ for shutdown cooling loops made inoperable by ICW. 3.4.6, "RCS Loops – MODE 4,"

LCO 3.7.8 (ICW) ~~3.7.4 At least~~ two ~~independent~~ ~~intake cooling water~~ loops shall be OPERABLE. LA01

Applicability **APPLICABILITY:** MODES 1, 2, 3, and 4.

ACTION:

ACTION A / ACTION A.1 ICW ~~With only~~ one ~~intake cooling water~~ loop in OPERABLE, restore ICW ~~at least two~~ loop~~s~~ to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following~~ 6 12 hours.

ACTION B.1, B.2 MODE 3

ACTION B.2 Note LCO 3.0.4.a is not applicable when entering ~~HOT SHUTDOWN~~. MODE 4 MODE 4

SURVEILLANCE REQUIREMENTS

ICW

~~4.7.4 At least~~ two ~~intake cooling water~~ loops shall be demonstrated OPERABLE:

ICW Add SR 3.7.8.1 Note A02

SR 3.7.8.1 ~~a.~~ In accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing that~~ each ~~valve (~~manual, power-operated~~, or~~ automatic~~)~~ servicing safety-related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

and valve in the flow path A03

ICW in the flow path

SR 3.7.8.2 ~~b.~~ In accordance with the Surveillance Frequency Control Program ~~during shutdown, by~~ verify~~ing that~~ each automatic valve servicing safety-related equipment actuates to its correct position on ~~a SIAS test~~ signal. LA02

the an actual or simulated actuation L01

that is not locked, sealed or otherwise secured in position L02

Add SR 3.7.8.3 M01

DISCUSSION OF CHANGES
ITS 3.7.8, INTAKE COOLING WATER (ICW) SYSTEM

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 4.7.4.1.a and Unit 2 4.7.4.a do not contain an explicit reference to isolating ICW flow to individual components. ITS SR 3.7.8.1 contains a Note which states, "Isolation of ICW flow to individual components does not render the ICW System inoperable." This changes CTS by adding an allowance that is not explicitly stated in the CTS. The purpose of the ICW System Technical Specification is to provide assurance that ICW is available to the appropriate plant components. This change is acceptable because by current use and application of the CTS, isolation of a component supplied with ICW does not necessarily result in the ICW System being considered inoperable, but the respective component may be declared inoperable for its system. This change clarifies this application. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 Unit 1 CTS 4.7.4.1.a and Unit 2 CTS 4.7.4.a require verification that each ICW valve (manual, power operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position. ITS SR 3.7.8.1 requires verification that each ICW manual, power operated, and automatic valve “in the flow path” servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in the correct position.

Unit 1 CTS 4.7.4.1.b and Unit 2 CTS 4.7.4.b require verification that each ICW automatic valve servicing safety related equipment actuates to it correct position on a safety injection actuation signal. ITS SR 3.7.8.2 requires verification that each ICW automatic valve “in the flow path” servicing safety related equipment that is not locked, sealed, or otherwise secured in position, actuates to its correct position on an actuation signal. This changes the CTS by adding the words "in the flow path" to Unit 1 CTS 4.7.4.1.a and Unit 2 CTS 4.7.4.a, and to Unit 1 CTS 4.7.4.1.b and Unit 2 CTS 4.7.4.b.

The purpose of Unit 1 CTS 4.7.4.1.a and Unit 2 CTS 4.7.4.a is to ensure all valves in the ICW System flow path are in the correct position. The purpose of Unit 1 CTS 4.7.4.1.b and Unit 2 CTS 4.7.4.b is to ensure all automatic valves in the ICW System flow path actuate to the correct position on an actual or simulated actuation signal. The addition of the words "in the flow path" does not change the intent of the Surveillance Requirements. Each ICW manual, power operated, and automatic valve servicing safety related equipment that is not locked, sealed, or otherwise secured in position will continue to be verified to be in the correct position. Each ICW automatic valve in the flow path that is not

locked, sealed or otherwise secured in position, will still be checked to ensure it actuates to the correct position on an actual or simulated actuation signal. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 4.7.4.1 and Unit 2 CTS 4.7.4 do not contain a requirement to verify each ICW System pump starts automatically on an actuation signal. ITS SR 3.7.8.3 states; "Verify each ICW pump starts automatically on an actual or simulated actuation signal," with a Frequency specified in accordance with the Surveillance Frequency Control Program. This changes the CTS by adding a Surveillance Requirement to test the ICW System pumps.

This change is necessary because in order for the ICW System to perform the safety function assumed in the accident analysis, the ICW pumps must start automatically. The ICW System is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. Therefore, the SR periodically verifies proper automatic operation of the pumps on an actual or simulated actuation signal. This Surveillance is similar to the testing requirements for other safety related pumps. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, SR 3.7.8.3 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 18 months consistent with the ISTS SR 3.7.8.3 and is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This change is designated as more restrictive because it adds a new SR.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.7.4.1 and Unit 2 CTS 3.7.4 states that two "independent" ICW loops shall be OPERABLE. ITS 3.7.8 requires two ICW loops to be OPERABLE but does not contain the detail that the loops must be independent. This changes the CTS by moving the detail that the ICW loops are independent to the Bases. The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for two ICW loops to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of

changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 4.7.4.1.b and Unit 2 CTS 4.7.4.b require verification of the automatic actuation of ICW components on a safety injection actuation signal (SIAS). ITS SR 3.7.8.2 does not state the specific type of signal, but only specifies an actual or simulated "actuation" signal. This changes CTS by moving the type of actuation signal (i.e., SIAS) to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to verify that appropriate equipment actuates upon receipt of an actuation signal. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5 of the ITS. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Specification.

LESS RESTRICTIVE CHANGES

L01 *(Category 6- Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.7.4.1.b and Unit 2 CTS 4.7.4.b require verification of the automatic actuation of intake cooling water components on a safety injection actuation signal (SIAS) "test" signal. ITS SR 3.7.8.2 specifies that the signal may be from either an "actual" or simulated (i.e., test) signal. This changes the CTS by explicitly allowing the use of either an actual or simulated signal for the test.

The purpose of Unit 1 CTS 4.7.4.1.b and Unit 2 CTS 4.7.4.b is to ensure the ICW components operate correctly upon receipt of an actuation signal. This change is acceptable because the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual," "simulated," or "test" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L02 *(Category 5 - Deletion of Surveillance Requirement)* Unit 1 CTS 4.7.4.1.b and Unit 2 CTS 4.7.4.b require verification that each ICW automatic valve in the flow path actuates to its correct position. ITS SR 3.7.8.2 requires verification that each ICW automatic valve in the flow path "that is not locked, sealed, or

otherwise secured in position" actuates to the correct position. This changes the CTS by excluding those valves that are locked, sealed, or otherwise secured in position from the verification.

The purpose of Unit 1 CTS 4.7.4.1.b and Unit 2 CTS 4.7.4.b is to provide assurance that if an event occurred requiring ICW valves to be in their correct position, then those requiring automatic actuation would actuate to their correct position. This change is acceptable because the deleted Surveillance is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested to in a manner and at a Frequency necessary to provide confidence that the equipment can perform its assumed safety function. Those automatic valves that are locked, sealed, or otherwise secured in position are not required to actuate on a safety injection actuation signal in order to perform their safety function because they are already in the required position. Testing such valves would not provide any additional assurance of OPERABILITY. Valves that are required to actuate will continue to be tested. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L03 **Unit 2 only:** *(Category 5 - Deletion of Surveillance Requirement)* CTS 3.7.4 note requires verification that when ICW pump 2C is being used to satisfy the requirements of this specification, the alignment of the discharge valves must be verified to be consistent with the appropriate power supply at least once per 24 hours. ITS 3.7.8 does not require this specific verification. This changes the CTS by deleting the specific verification of valve and power supply alignment when the swing ICW pump is being used.

The purpose of the CTS 3.7.4 Note is to ensure that when the swing ICW pump (Pump 2C) is used to satisfy the requirements, the proper valve and power supply alignment is verified periodically. The PSL ICW System design includes three pumps: each pump provides flow to a redundant ICW loop. These two pumps are powered from the associated essential bus. The third swing pump can be powered from either essential bus and the discharge valves can be aligned to either ICW loop. When used to support ICW loop OPERABILITY, the swing pump power and valves must be aligned to the correct ICW loop. The Surveillances retained in ITS will continue to ensure the valves and breakers are properly aligned when the swing pump is being used. ITS SR 3.7.8.1 requires verification that the valves that are not locked sealed or otherwise secured in position (i.e., accident position) are in the correct position. This includes verifying the discharge valves of the swing pump are in their correct position to support ICW loop OPERABILITY. ITS SR 3.7.8.3 requires verification that the required pump automatically starts on an actual or simulated actuation signal. This test ensures the swing pump is powered from the correct essential bus by verifying the pump can automatically start on the selected ICW loop. The Frequency in accordance with the Surveillance Frequency Control Program (i.e., 31 days for the valve position verification and 18 months for the automatic pump start) has been determined a sufficient periodicity for verification of required equipment. This change is acceptable because the deleted Surveillance Requirement is not necessary to ensure that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a

manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. This change is designated as less restrictive because a Surveillance which is required in the CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7.8 ~~Service Water System (SWS)~~ ← Intake Cooling Water (ICW) System

3.7.4.1 LCO 3.7.8 Two ~~SWS trains~~ (ICW loops) shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.4.1 Action A. One ~~SWS train~~ (ICW loop) inoperable.	A.1 ----------NOTE~~S~~---------- ~~1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by SWS.~~	
3.7.4.1 Action Note	~~2.~~ Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for shutdown cooling made inoperable by ~~SWS~~ (ICW). ----------	
3.7.4.1 Action	Restore ~~SWS train~~ (ICW loop) to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.4.1 Action B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.7.4.1.a DOC A02 SR 3.7.8.1	ICW ----------NOTE---------- Isolation of ~~SWS~~ flow to individual components does not render the ~~SWS~~ inoperable. ---------- ICW	1 1
4.7.4.1.a	Verify each ~~SWS~~ manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.7.4.1.b	SR 3.7.8.2	Verify each ~~SWS~~ (ICW) automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ 1 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2
DOC M01	SR 3.7.8.3	Verify each ~~SWS~~ (ICW) pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ 1 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

3.7 PLANT SYSTEMS

3.7.8 ~~Service Water System (SWS)~~ ← Intake Cooling Water (ICW) System

3.7.4 LCO 3.7.8 Two ~~SWS trains~~ (ICW loops) shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.4 Action A. One ~~SWS train~~ (ICW loop) inoperable.	A.1 ----------NOTE~~S~~---------- ~~1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by SWS.~~ 3.7.4 Action Note ~~2.~~ Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for shutdown cooling made inoperable by ~~SWS~~. ← ICW ---------- ICW loop 3.7.4 Action Restore ~~SWS train~~ to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.4 Action B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.7.4.a DOC A02 SR 3.7.8.1	ICW ----------NOTE---------- Isolation of ~~SWS~~ flow to individual components does not render the ~~SWS~~ inoperable. ---------- ICW	1 1
4.7.4.a	Verify each ~~SWS~~ manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.7.4.b	SR 3.7.8.2	Verify each ~~SWS~~ automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[[18] months~~ 1 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2
DOC M01	SR 3.7.8.3	Verify each ~~SWS~~ pump starts automatically on an actual or simulated actuation signal.	~~[[18] months~~ 1 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.8, INTAKE COOLING WATER (ICW) SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. PSL Unit 1 and Unit 2 diesel generators are designed with a self-contained cooling system and do not depend on any support plant cooling water system. Therefore, Note 1 to ISTS 3.7.8 is unnecessary and deleted. Concomitant changes made to reflect the Note deletion.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.8 ~~Service Water System (SWS)~~ Intake Cooling Water (ICW) System

BASES

BACKGROUND

The ~~SWS~~ provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation or a normal shutdown, the ~~SWS~~ also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The ~~SWS~~ consists of two ~~separate~~, 100% capacity safety related cooling water ~~trains~~. Each ~~train~~ consists of ~~two~~ 100% capacity pump~~s~~, one component cooling water (CCW) heat exchanger, piping, valves, instrumentation, and ~~two cyclone separators~~. The pumps and valves are remote manually aligned, except in the unlikely event of a loss of coolant accident (LOCA). The pumps aligned to the critical loops are automatically started upon receipt of a safety injection actuation signal and all essential valves are aligned to their post accident positions. ~~The SWS also provides emergency makeup to the spent fuel pool and CCW System [and is the backup water supply to the Auxiliary Feedwater System]~~.

Additional information about the design and operation of the ~~SWS~~, along with a list of the components served, is presented in the FSAR, Section ~~[~~9.2.1~~]~~ (Ref. 1). The principal safety related function of the ~~SWS~~ is the removal of decay heat from the reactor via the ~~[~~CCW System~~]~~.

APPLICABLE SAFETY ANALYSES

The design basis of the ~~SWS~~ is for one ~~SWS train~~, in conjunction with the CCW System and a 100% capacity containment cooling system (containment spray, containment coolers, or a combination), removing core decay heat 20 minutes following a design basis LOCA, as discussed in the FSAR, Section ~~[~~6.2~~]~~ (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the safety injection pumps. The ~~SWS~~ is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The ~~SWS~~, in conjunction with the CCW System, also cools the unit from shutdown cooling (SDC), as discussed in the FSAR, Section ~~[5.4.7]~~ (Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the

ICW System
ICW System
Three ICW pumps are provided. A pump can be removed from service without reducing the capability or redundancy of the ICW System. Two of the ICW pumps are powered from the associated essential bus. The third ICW pump can be powered from either essential bus.
ICW System
loops
loop
independent
one
strainer
ICW System
U
ICW System
ICW loop
U
ICW System
9.2.2
U
All nonessential components are isolated. ICW flows through a Debris Filter System (DFS) at the inlet of each CCW heat exchanger. The DFS backwash automatically through a debris discharge valve and reenter the ICW System downstream of the ICW discharge valves. Automatic backwashing is not a safety-related function. During a DBA, the debris discharge valve is signaled to close by the safety injection actuation signal to prevent flow diversion past the heat exchanger.

1
2
1
R1
2
1
2
1
2
1

BASES

APPLICABLE SAFETY ANALYSES (continued)

number of CCW and SDC System trains that are operating. One ~~SWS train~~ is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes that a maximum ~~SWS~~ temperature of 95°F occurring simultaneously with maximum heat loads on the system.

The ~~SWS~~ satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two ~~SWS trains~~ are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power.

An ~~SWS train~~ is considered OPERABLE when:

a. The associated pump is OPERABLE and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

The spare ICW pump can be substituted for a normal ICW pump, provided the spare ICW valves are aligned to the applicable loop and the power supply for the spare pump is aligned to the same essential bus as the pump it is replacing.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ~~SWS~~ System is a normally operating system, which is required to support the OPERABILITY of the equipment serviced by the ~~SWS~~ and required to be OPERABLE in these MODES.

Although the LCO for the ~~SWS~~ is not applicable in MODES 5 and 6, the capability of the ~~SWS~~ to perform its necessary related support functions may be required for OPERABILITY of supported systems.

ACTIONS

A.1

With one ~~SSW train~~ inoperable, action must be taken to restore OPERABLE status within 72 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. In this Condition, the remaining OPERABLE ~~SWS train~~ is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the ~~SWS train~~ could result in loss of ~~SWS~~ function. Required Action A.1 is modified by ~~two~~ Note~~s~~. ~~The first Note indicates that the applicable Conditions of LCO 3.8.1, "AC Sources - Operating," should be entered if the inoperable SWS train results in an inoperable emergency diesel generator.~~ The ~~second~~ Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ~~SWS train~~ results in an inoperable SDC. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE ~~train~~, and the low probability of a DBA occurring during this time period.

BASES

ACTIONS (continued)

B.1 and B.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

2

ICW loop

If the ~~SWS train~~ cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is minimized. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

1

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual, power operated, and automatic valves in the ~~SWS~~ flow path ensures that the proper flow paths exist for ~~SWS~~ operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of the ~~SWS~~ components or systems may render those components inoperable but does not affect the OPERABILITY of the ~~SWS~~.

ICW

ICW

ICW System

1

1

[~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.7.8.2

in the flow path servicing safety related equipment

ICW System

ICW

1

1

This SR verifies proper automatic operation of the ~~SWS~~ valves on an actual or simulated actuation signal. The ~~SWS~~ is a normally operating system that cannot be fully actuated as part of the normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. [~~The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

2

~~OR~~

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.7.8.3

ICW System ICW

The SR verifies proper automatic operation of the ~~SWS~~ pumps on an actual or simulated actuation signal. The ~~SWS~~ is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

REFERENCES

1. FSAR, Section ~~[~~9.2.1~~]~~. U
2. FSAR, Section ~~[~~6.2~~]~~. U
3. FSAR, Section ~~[5.4.7]~~. U 9.2.2
4. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

1 2
1 2
1 2

1

B 3.7 PLANT SYSTEMS

B 3.7.8 ~~Service Water System (SWS)~~ Intake Cooling Water (ICW)

BASES

BACKGROUND

The ~~SWS~~ provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation or a normal shutdown, the ~~SWS~~ also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The ~~SWS~~ consists of two ~~separate~~, 100% capacity safety related cooling water ~~trains~~. Each ~~train~~ consists of ~~two~~ 100% capacity pump~~s~~, one component cooling water (CCW) heat exchanger, piping, valves, instrumentation, and ~~two cyclone separators~~. The pumps and valves are remote manually aligned, except in the unlikely event of a loss of coolant accident (LOCA). The pumps aligned to the critical loops are automatically started upon receipt of a safety injection actuation signal and all essential valves are aligned to their post accident positions. ~~The SWS also provides emergency makeup to the spent fuel pool and CCW System [and is the backup water supply to the Auxiliary Feedwater System]~~.

Additional information about the design and operation of the ~~SWS~~, along with a list of the components served, is presented in the FSAR, Section ~~[~~9.2.1~~]~~ (Ref. 1). The principal safety related function of the ~~SWS~~ is the removal of decay heat from the reactor via the ~~[~~CCW System~~]~~.

APPLICABLE SAFETY ANALYSES

The design basis of the ~~SWS~~ is for one ~~SWS train~~, in conjunction with the CCW System and a 100% capacity containment cooling system (containment spray, containment coolers, or a combination), removing core decay heat 20 minutes following a design basis LOCA, as discussed in the FSAR, Section ~~[~~6.2~~]~~ (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the safety injection pumps. The ~~SWS~~ is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The ~~SWS~~, in conjunction with the CCW System, also cools the unit from shutdown cooling (SDC), as discussed in the FSAR, Section ~~[5.4.7]~~ (Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the

Three ICW pumps are provided. A pump can be removed from service without reducing the capability or redundancy of the ICW System. Two of the ICW pumps are powered from the associated essential bus. The third ICW pump can be powered from either essential bus.

All nonessential components are isolated. ICW flows through a Debris Filter System (DFS) at the inlet of each CCW heat exchanger. The DFS backwash automatically through a debris discharge valve and reenter the ICW System downstream of the ICW discharge valves. Automatic backwashing is not a safety-related function. During a DBA, the debris discharge valve is signaled to close by the safety injection actuation signal to prevent flow diversion past the heat exchanger.

BASES

APPLICABLE SAFETY ANALYSES (continued)

number of CCW and SDC System trains that are operating. One ~~SWS train~~ is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes that a maximum ~~SWS~~ temperature of 95°F occurring simultaneously with maximum heat loads on the system.

The ~~SWS~~ satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two ~~SWS trains~~ are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power.

An ~~SWS train~~ is considered OPERABLE when:

a. The associated pump is OPERABLE and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

The spare ICW pump can be substituted for a normal ICW pump, provided the spare ICW valves are aligned to the applicable loop and the power supply for the spare pump is aligned to the same essential bus as the pump it is replacing.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ~~SWS~~ System is a normally operating system, which is required to support the OPERABILITY of the equipment serviced by the ~~SWS~~ and required to be OPERABLE in these MODES.

Although the LCO for the ~~SWS~~ is not applicable in MODES 5 and 6, the capability of the ~~SWS~~ to perform its necessary related support functions may be required for OPERABILITY of supported systems.

ACTIONS

A.1

With one ~~SSW train~~ inoperable, action must be taken to restore OPERABLE status within 72 hours ~~[~~or in accordance with the Risk Informed Completion Time Program~~]~~. In this Condition, the remaining OPERABLE ~~SWS train~~ is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the ~~SWS train~~ could result in loss of ~~SWS~~ function. Required Action A.1 is modified by ~~two~~ Note~~s~~. ~~The first Note indicates that the applicable Conditions of LCO 3.8.1, "AC Sources - Operating," should be entered if the inoperable SWS train results in an inoperable emergency diesel generator.~~ The ~~second~~ Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ~~SWS train~~ results in an inoperable SDC. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE ~~train~~, and the low probability of a DBA occurring during this time period.

BASES

ACTIONS (continued)

B.1 and B.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

2

ICW loop

If the ~~SWS train~~ cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is minimized. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

1

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual, power operated, and automatic valves in the ~~SWS~~ flow path ensures that the proper flow paths exist for ~~SWS~~ operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of the ~~SWS~~ components or systems may render those components inoperable but does not affect the OPERABILITY of the ~~SWS~~.

ICW

ICW

ICW System

1

1

[~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

2

SR 3.7.8.2

in the flow path servicing safety related equipment

ICW System

ICW

1

1

This SR verifies proper automatic operation of the ~~SWS~~ valves on an actual or simulated actuation signal. The ~~SWS~~ is a normally operating system that cannot be fully actuated as part of the normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. [~~The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

2

~~OR~~

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

SR 3.7.8.3

ICW System ICW

The SR verifies proper automatic operation of the ~~SWS~~ pumps on an actual or simulated actuation signal. The ~~SWS~~ is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. ~~[The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ 1 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

REFERENCES

1. FSAR, Section ~~[~~9.2.1~~]~~. U 1 2
2. FSAR, Section ~~[~~6.2~~]~~. U 1 2
3. FSAR, Section ~~[5.4.7]~~. 9.2.2 U 1 2
4. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.8, BASES, INTAKE COOLING WATER (ICW) SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. PSL Unit 1 and Unit 2 diesel generators are designed with a self-contained cooling system and do not depend on any support plant cooling water system. Therefore, Note 1 to ISTS 3.7.8 is unnecessary and deleted. Concomitant changes made to reflect the Note deletion.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.8, INTAKE COOLING WATER (ICW) SYSTEM

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 9

3.7.9, Ultimate Heat Sink (UHS)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

PLANT SYSTEMS

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

UHS

LCO 3.7.9 ~~3.7.5.1~~ The ~~ultimate heat sink~~ shall be OPERABLE ~~with:~~

~~a.~~ ~~Cooling water from the Atlantic Ocean~~ ~~providing a~~ water level ~~above~~ LA01

SR 3.7.9.1 of UHS is > –10.5 feet elevation, Mean Low Water~~, at the plant intake structure, and~~

~~b.~~ ~~Two OPERABLE valves in the barrier dam between Big Mud Creek and the intake structure.~~ LA02

Applicability **APPLICABILITY**: ~~At all times~~. MODES 1, 2, 3, and 4 L01

ACTION:

MODE 3

ACTION A ~~a.~~ With the water level requirement of the above Specification not satisfied, be in ~~at least HOT STANDBY~~ within six hours and provide cooling water from ~~Big Mud Creek~~ within ~~the next~~ ~~12~~ hours.

backup supply 18

~~b.~~ ~~With one isolation valve in the barrier dam between Big Mud Creek and the intake structure inoperable, restore the inoperable valve to OPERABLE status within 72 hours or, within the next 24 hours, install a temporary flow barrier and open the barrier dam isolation valve. The availability of the onsite equipment capable of removing the barrier shall be verified at least once per seven days thereafter.~~

~~c.~~ ~~With both of the isolation valves in the barrier dam between the intake structure and Big Mud Creek inoperable, within 24 hours either:~~ LA02

~~1) Install both temporary flow barriers and manually open both barrier dam isolation valves. The availability of the onsite equipment capable of removing the barriers shall be verified at least once per seven days thereafter, or~~

~~2) Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

ACTION B Add ACTION B M01

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1 ~~4.7.5.1.1~~ ~~The ultimate heat sink shall be determined OPERABLE~~ in accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing the average~~ water level ~~to be within the limits~~.

~~4.7.5.1.2~~ ~~The isolation valves in the barrier dam between the intake structure and Big Mud Creek shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by cycling each valve through at least one complete cycle of full travel.~~ LA02

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PLANT SYSTEMS

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

LCO 3.7.9

UHS

3.7.5.1 The ~~ultimate heat sink~~ shall be OPERABLE ~~with:~~

~~a.~~ ~~Cooling water from the Atlantic Ocean~~ ~~providing a~~ water level ~~above~~ –10.5 feet elevation, Mean Low Water~~,~~ ~~at the plant intake structure,~~ and — LA01

SR 3.7.9.1 — of UHS is >

~~b.~~ ~~Two OPERABLE valves in the barrier dam between Big Mud Creek and the intake structure.~~ — LA02

Applicability

APPLICABILITY: ~~At all times~~. — MODES 1, 2, 3, and 4 — L01

ACTION:

ACTION A

MODE 3

~~a.~~ With the water level requirement of the above specification not satisfied, be in ~~at least HOT STANDBY~~ within 6 hours and provide cooling water from ~~Big Mud Creek~~ within ~~the next 12~~ hours.

backup supply — 18

~~b.~~ ~~With one isolation valve in the barrier dam between Big Mud Creek and the intake structure inoperable, restore the inoperable valve to OPERABLE status within 72 hours, or within the next 24 hours, install a temporary flow barrier and open the barrier dam isolation valve. The availability of the onsite equipment capable of removing the barrier shall be verified at least once per 7 days thereafter.~~

~~c.~~ ~~With both of the isolation valves in the barrier dam between the intake structure and Big Mud Creek inoperable, within 24 hours, either:~~ — LA02

~~1. Install both temporary flow barriers and manually open both barrier dam isolation valves. The availability of the onsite equipment capable of removing the barriers shall be verified at least once per 7 days thereafter, or~~

~~2. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

ACTION B

Add ACTION B — M01

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

~~4.7.5.1.1 The ultimate heat sink shall be determined OPERABLE~~ in accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing the average~~ water level ~~to be within the limits~~.

~~4.7.5.1.2 The isolation valves in the barrier dam between the intake structure and Big Mud Creek shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by cycling each valve through at least one complete cycle of full travel.~~ — LA02

DISCUSSION OF CHANGES
ITS 3.7.9, ULTIMATE HEAT SINK (UHS)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 3.7.5.1 does not provide any actions when the UHS water level is not OPERABLE, and either action is not completed within its completion time. Therefore, CTS 3.0.3 would be entered requiring entry into HOT STANDBY (MODE 3) within 7 hours and COLD SHUTDOWN (MODE 5) within 37 hours. ITS 3.7.9 ACTION B, which applies when the UHS in inoperable (i.e., water level is not within limit) and a Required Action and associated Completion Time is not met, requires entry into COLD SHUTDOWN (MODE 5) within 36 hours. ITS 3.7.9, Required Action A.1 required entry into MODE 3 within 6 hours, therefore, this requirement is not repeated in ACTION B. This changes the CTS by providing one less hour to shut down the unit to MODE 5 when the UHS water level is inoperable, and a Required Action and associated Completion Time is not met.

The purpose of requiring a shutdown when Required Action A.1 or Required Action A.2 is not completed within its associated Completion Time, is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 5 within 36 hours. This change is acceptable because it provides an adequate period of time to be in a MODE in which the requirement does not apply, commensurate with the severity of the inoperability. The Completion Time of 36 hours is reasonable, based on operating experience, for reaching MODE 5, from full power in an orderly manner and without challenging unit systems. The action to be in MODE 3 within 6 hours is provided in ACTION A. This change has been designated as more restrictive because it reduces the Completion Times to be in MODE 5.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.7.5.1.a states that the ultimate heat sink (UHS) shall be OPERABLE with cooling water from the Atlantic Ocean providing a water level above –10.5 feet elevation, Mean Low Water, at the plant intake structure. ITS

3.7.9 states that the UHS shall be OPERABLE. This changes the CTS by removing design details of the UHS water source (i.e., Atlantic Ocean, and plant intake structure from which UHS water level is determined).

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the UHS be OPERABLE and requires verification of the UHS water level at the Frequency specified in the Surveillance Frequency Control Program. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 4 - Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPM, IST Program, or ISI Program)* CTS 3.7.5.1.b requires two OPERABLE isolation valves in the barrier dam between Big Mud Creek and the intake structure. CTS 3.7.5.1 Action b provides actions with one isolation valve in the barrier dam inoperable. CTS 3.7.5.1 Action c provides actions with both isolation valves in the barrier dam inoperable. Additionally, CTS 4.7.5.1.2 demonstrates the isolation valves are OPERABLE by cycling each valve through at least one complete cycle of full travel. These LCO, Action and Surveillance requirements are not included in the ITS. This changes the CTS by moving the explicit ultimate heat sink barrier dam valve requirements from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details from the Technical Specification is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. The purpose of the valves in the barrier dam between Big Mud Creek and the intake structure is to provide an alternate intake structure fill system to support safe shutdown in the event the primary method of filling the intake structure (i.e., Atlantic Ocean) becomes unavailable following a design basis seismic event. These valves are not credited in the mitigation of any design basis accident or transient specified in Chapter 15 of the UFSAR.

This change is acceptable because the LCO requirements continue to ensure that the primary source of water to the ultimate heat sink (UHS) is OPERABLE consistent with the safety analyses and licensing basis. The UHS requirements associated with the valves in the barrier dam between Big Mud Creek and the intake structure are not part of a primary success path in the mitigation of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Additionally, these requirements are not discussed in the PSL probabilistic risk assessment and do not constitute a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Therefore, these CTS requirements do not meet the applicable 10 CFR 50.36(c)(2)(ii) criteria for retention in the Technical Specifications.

Also, this change is acceptable because the removed information will be adequately controlled in the TRM. Changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a requirement is being removed from the Technical Specifications. | R1

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.7.5.1 Applicability is "at all times." ITS LCO 3.7.9 Applicability is MODES 1, 2, 3, and 4. This changes the CTS by only requiring the UHS to be OPERABLE in MODES 1, 2, 3, and 4.

The purpose of the CTS 3.7.5.1 Applicability is to specify the MODES and other specified conditions in which the LCO must be met. This change is acceptable because the requirements continue to ensure the UHS is maintained OPERABLE in the MODES assumed in the safety analyses and licensing basis. In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES. Although the LCO for the UHS is not applicable in MODES 5 and 6, the capability of the UHS to perform its necessary related support functions may be required for OPERABILITY of supported systems. UHS support requirements for systems required in MODES 5 and 6 are not the same as for the systems required in MODES 1, 2, 3, and 4 and the heat load requirements in the shutdown MODES is less. Therefore, the definition of OPERABILITY will ensure the UHS requirements necessary to support systems required in MODES 5 and 6 will not be impacted. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

3.7.5.1 LCO 3.7.9 The UHS shall be OPERABLE.

DOC L01 APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. [One or more cooling towers with one cooling tower fan inoperable.~~	~~A.1 Restore cooling tower fan(s) to OPERABLE status.~~	~~7 days~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~
3.7.5.1 Action a. ~~B~~. (A) ~~Required Action and associated Completion Time of Condition A or B not met~~. UHS not within limit	~~B~~.1 (A) Be in MODE 3. AND ~~B~~.2 (A) ~~NOTE~~ ~~LCO 3.0.4.a is not applicable when entering MODE 4.~~ ~~Be in MODE 4~~. Provide cooling water from backup supply	6 hours ~~12~~ hours (18)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~---REVIEWER'S NOTE---~~ ~~The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit.~~ ~~---~~ ~~C. [Water temperature of the UHS > [90]°F and ≤ [ ]°F.~~	~~C.1~~ ~~Verify water temperature of the UHS is ≤ [90]°F averaged over the previous 24 hour period.~~	~~Once per hour]~~ 6 2
~~D.~~ B [Required Action and associated Completion Time ~~of Condition C~~ not met. ~~OR]~~ ~~UHS inoperable [for reasons other than Condition A or C].~~	~~D.1~~ ~~Be in MODE 3.~~ ~~AND~~ ~~D.2~~ B.1 Be in MODE 5.	~~6 hours~~ 2 3 5 36 hours 3 2

DOC M01

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.7.5.1.1 SR 3.7.9.1	~~[~~ Verify water level of UHS is ≥ ~~[562]~~ > -10.5 ft ~~[mean sea level]~~ mean low water level.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~ 2 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.9.2	[Verify average water temperature of UHS is ≤ [90]°F.	[24 hours OR In accordance with the Surveillance Frequency Control Program]]
SR 3.7.9.3	[Operate each cooling tower fan for ≥ [15] minutes.	[31 days OR In accordance with the Surveillance Frequency Control Program]]

6

4

1

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

3.7.5.1 LCO 3.7.9 The UHS shall be OPERABLE.

DOC L01 APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
~~A. [One or more cooling towers with one cooling tower fan inoperable.~~	~~A.1~~ ~~Restore cooling tower fan(s) to OPERABLE status.~~	~~7 days~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~	4
3.7.5.1 Action a. ~~B~~. (A) ~~Required Action and associated Completion Time of Condition A or B not met~~. UHS not within limit	~~B~~.1 (A) Be in MODE 3. AND ~~B~~.2 (A) ~~NOTE LCO 3.0.4.a is not applicable when entering MODE 4.~~ ~~Be in MODE 4~~. Provide cooling water from backup supply	6 hours ~~12~~ hours (18)	3 1 3 4

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~---REVIEWER'S NOTE---~~ ~~The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit.~~ ~~---~~ ~~C. [Water temperature of the UHS > [90]°F and ≤ [ ]°F.~~	~~C.1~~ ~~Verify water temperature of the UHS is ≤ [90]°F averaged over the previous 24 hour period.~~	~~Once per hour]~~
~~D.~~ ~~[~~ Required Action and associated Completion Time ~~of Condition C~~ not met. ~~OR]~~ ~~UHS inoperable [for reasons other than Condition A or C].~~	~~D.1~~ ~~Be in MODE 3.~~ ~~AND~~ ~~D.2~~ Be in MODE 5.	~~6 hours~~ 36 hours

DOC M01

B

B.1

6 2 3 5 3 2 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	~~[~~ Verify water level of UHS is ~~≥ [562]~~ ft ~~[mean sea level]~~.	~~[24 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~

4.7.5.1.1

> -10.5

mean low water level

2 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.7.9.2~~	~~[Verify average water temperature of UHS is ≤ [90]°F.~~	~~[24 hours~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~
~~SR 3.7.9.3~~	~~[Operate each cooling tower fan for ≥ [15] minutes.~~	~~[31 days~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

6

4

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.9, ULTIMATE HEAT SINK (UHS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS ACTIONS numbering.

4. The PSL UHS design does not include cooling towers, therefore, ISTS 3.7.9 ACTION A and ISTS SR 3.7.9.3 are deleted.

5. ISTS 3.7.9 ACTION D (ITS ACTION B) is added to be in MODE 5 within 36 hours with the Required Action and associated Completion Time (i.e., of ITS Condition A) not met. The ISTS Required Action D.1 to be in MODE 3 within 6 hours is not included in the ITS because ITS 3.7.9, Required Action A.1 requires the unit to be in MODE 3 within 6 hours if the UHS is inoperable. See CTS Discussion of Change M01 for the discussion regarding the addition of ITS 3.7.9 ACTION B.

6. The PSL Current Technical Specifications do not include UHS water temperature. The ICW System was designed to remove the post-accident heat load from the CCW system, based on seawater temperature determined from historical data and Atlantic Ocean temperature surveys. To assure the operability of the CCW system, the maximum design seawater (ICW) inlet temperature is 95°F. Historical data and Atlantic Ocean temperature studies show maximum Atlantic Ocean seawater temperature at 87°F and represents ambient conditions at the St Lucie Plant in the Environmental Report. Therefore, ISTS 3.7.9 ACTION C and ISTS SR 3.7.9.2 are not included in the ITS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for process and operating heat from safety related components during a Design Basis Accident (DBA) or transient, as well as during normal operation. This is done utilizing the ~~Service Water~~ System.

Intake Cooling Water (ICW)

1

Insert 1

~~The UHS has been defined as that complex of water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as, discussed in the FSAR, Section [9.2.5] (Ref. 1). If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink.~~ The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

1

~~A variety of complexes is used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required.~~

1

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. ~~Basins of cooling towers generally include less than a 30 day supply of water, typically 7 days or less. A 30 day supply would be dependent on another source(s) and a makeup system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a 1 day supply, the systems for replenishing the basin and the backup source(s) become of sufficient importance that the makeup system itself may be required to meet the same design criteria as an Engineered Safety Feature (e.g., single failure considerations, and multiple makeup water sources may be required).~~

1

~~It follows that the many variations in the UHS configurations will result in many unit to unit variations in OPERABILITY determinations and SRs. The ACTIONS and SRs are illustrative of a cooling tower UHS without a makeup requirement.~~

1

Additional information on the design and operation of the system along with a list of components served can be found in Reference 1.

1

INSERT 1

The UHS supplies water to the ICW System to perform its intended function, as discussed in the UFSAR, Section 9.2.7 (Ref. 1). There are two independent water sources and a discharge canal for the UHS. The primary water source is the Atlantic Ocean through the intake canal. This water source is used for all plant normal and accident conditions. The second source (backup supply) is Big Mud Creek which is fed by the Indian River (ultimately the Atlantic Ocean). The backup supply is utilized in the event the intake canal is rendered unable to supply sufficient water, as indicated by UHS intake level, due to a safe shutdown earthquake. Regardless of source, the cooling water is discharged into the discharge canal and flows through the discharge canal to the Atlantic Ocean. To ensure a supply of water for safe shutdown or accident conditions following a design basis seismic event, the backup supply water flow from Big Mud Creek is initiated from either unit's control room whenever the intake canal level falls below -10.5 feet. The minimum required intake canal level provides adequate margin above the minimum level for continued operation of the ICW pumps.

BASES

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on shutdown cooling. ~~For those units using it as the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load. Its maximum post accident heat load occurs 20 minutes after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation, and the containment cooling systems are required to remove the core decay heat.~~ 1

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis. ~~The assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case failure (e.g., single failure of a manmade structure).~~ 1 The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the UHS.

The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water ~~at or below the maximum temperature~~ that would allow the ~~SWS~~ ICW System to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the ~~SWS~~ ICW System. To meet this condition, the UHS ~~temperature should not exceed [90]°F and the~~ level should not fall below ~~[562 ft mean sea~~ -10.5 feet mean low water ~~level]~~ during normal unit operation. 1 R1 1 R1 2

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

Although the LCO for the UHS is not applicable in MODES 5 and 6, the capability of the UHS to perform its necessary related support functions may be required for OPERABILITY of supported systems.

ACTIONS

~~[A.1~~

~~If one or more cooling towers have one fan inoperable (i.e., at least one fan per cooling tower is OPERABLE), action must be taken to restore the inoperable cooling tower fan(s) to OPERABLE status within 7 days [or in accordance with the Risk Informed Completion Time Program].~~ 4

BASES

ACTIONS (continued)

~~The 7 day Completion Time is reasonable, based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable, the number of available systems, and the time required to complete the action.]~~ 4

A A

~~B~~.1 and ~~B~~.2 1

Insert 2

~~----------REVIEWER'S NOTE----------~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

~~----------~~

~~If the cooling tower fan cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours.~~

~~Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 3). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.~~

~~Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and~~

1

INSERT 2

With the UHS inoperable, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to MODE 3 and the backup cooling water supply must be aligned to the intake canal. Risk in MODE 3 is minimized by providing makeup from a backup water source to the intake canal. This action ensures the backup supply is aligned to restore cooling water for the ICW pumps if the primary supply cannot be restored to within limit. Also, there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms in MODES 3 and 4 than in MODE 5. Voluntary entry into MODE 4 or 5 may be made as these MODES are also acceptable low-risk states. Required Action A.2 is provided to supply cooling water from the backup water supply within 18 hours. The Completion Time for Required Action A.2 allows the operator time to evaluate and repair any discovered inoperabilities while providing adequate time to align the backup supply to the intake canal. If the backup water supply cannot be established within the required Completion Time, Condition B must be entered.

BASES

ACTIONS (continued)

establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power in an orderly manner and without challenging plant systems.

[C.1

REVIEWER'S NOTE

The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit.

With water temperature of the UHS > [90]°F, the design basis assumption associated with initial UHS temperature are bounded provided the temperature of the UHS averaged over the previous 24 hour period is ≤ [90]°F. With the water temperature of the UHS > [90]°F, long term cooling capability of the ECCS loads and DGs may be affected. Therefore, to ensure long term cooling capability is provided to the ECCS loads when water temperature of the UHS is > [90]°F, Required Action C.1 is provided to more frequently monitor the water temperature of the UHS and verify the temperature is ≤ [90]°F when averaged over the previous 24 hour period. The once per hour Completion Time takes into consideration UHS temperature variations and the increased monitoring frequency needed to ensure design basis assumptions and equipment limitations are not exceeded in this condition. If the water temperature of the UHS exceeds [90]°F when averaged over the previous 24 hour period or the water temperature of the UHS exceeds [ ]°F, Condition D must be entered immediately.] 5

B

[D.1 and D.2 3

A are

If the Required Actions or Completion Times of Condition C is not met, or the UHS is inoperable [for reasons other than Condition A or C], the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.] 2 1

is

BASES

SURVEILLANCE REQUIREMENTS

~~[~~ SR 3.7.9.1

This SR verifies adequate long term (30 days) cooling can be maintained. The level specified also ensures sufficient NPSH is available for operating the ~~SWS~~ pumps. ~~[The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ [562] ft [mean sea level].~~

ICW

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]]~~

2

~~[SR 3.7.9.2~~

~~This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature within the maximum accident or normal design heat loads for 30 days following a DBA. [The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water temperature is ≤ [92]°F.~~

5

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.7.9.3

Operating each cooling tower fan for ≥ [15] minutes verifies that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration can be detected for corrective action. [The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances. 4

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]]

U 1

REFERENCES 1. FSAR, Section [9.2.5]. 7 2

2. Regulatory Guide 1.27.

3. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001. 1

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for process and operating heat from safety related components during a Design Basis Accident (DBA) or transient, as well as during normal operation. This is done utilizing the ~~Service Water~~ System.

Intake Cooling Water (ICW)

1

Insert 1

~~The UHS has been defined as that complex of water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as, discussed in the FSAR, Section [9.2.5] (Ref. 1). If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink.~~ The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

1

~~A variety of complexes is used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required.~~

1

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. ~~Basins of cooling towers generally include less than a 30 day supply of water, typically 7 days or less. A 30 day supply would be dependent on another source(s) and a makeup system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a 1 day supply, the systems for replenishing the basin and the backup source(s) become of sufficient importance that the makeup system itself may be required to meet the same design criteria as an Engineered Safety Feature (e.g., single failure considerations, and multiple makeup water sources may be required).~~

1

~~It follows that the many variations in the UHS configurations will result in many unit to unit variations in OPERABILITY determinations and SRs. The ACTIONS and SRs are illustrative of a cooling tower UHS without a makeup requirement.~~

1

Additional information on the design and operation of the system along with a list of components served can be found in Reference 1.

1

1

INSERT 1

The UHS supplies water to the ICW System to perform its intended function, as discussed in the UFSAR, Section 9.2.5 (Ref. 1). There are two independent water sources and a discharge canal for the UHS. The primary water source is the Atlantic Ocean through the intake canal. This water source is used for all plant normal and accident conditions. The second source (backup supply) is Big Mud Creek which is fed by the Indian River (ultimately the Atlantic Ocean). The backup supply is utilized in the event the intake canal is rendered unable to supply sufficient water, as indicated by UHS intake level, due to a safe shutdown earthquake. Regardless of source, the cooling water is discharged into the discharge canal and flows through the discharge canal to the Atlantic Ocean. To ensure a supply of water for safe shutdown or accident conditions following a design basis seismic event, the backup supply water flow from Big Mud Creek is initiated from either unit's control room whenever the intake canal level falls below -10.5 feet. The minimum required intake canal level provides adequate margin above the minimum level for continued operation of the ICW pumps.

BASES

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on shutdown cooling. ~~For those units using it as the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load. Its maximum post accident heat load occurs 20 minutes after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation, and the containment cooling systems are required to remove the core decay heat.~~ 1

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis. ~~The assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case failure (e.g., single failure of a manmade structure).~~ 1 The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the UHS.

The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water ~~at or below the maximum temperature~~ that would allow the ~~SWS~~ ICW System to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the ~~SWS~~ ICW System. To meet this condition, the UHS ~~temperature should not exceed [90]°F and the~~ level should not fall below ~~[562 ft mean sea~~ -10.5 feet mean low water level~~]~~ during normal unit operation. 1 R1 1 R1 2

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

Although the LCO for the UHS is not applicable in MODES 5 and 6, the capability of the UHS to perform its necessary related support functions may be required for OPERABILITY of supported systems.

ACTIONS

~~[A.1~~

~~If one or more cooling towers have one fan inoperable (i.e., at least one fan per cooling tower is OPERABLE), action must be taken to restore the inoperable cooling tower fan(s) to OPERABLE status within 7 days [or in accordance with the Risk Informed Completion Time Program].~~ 4

BASES

ACTIONS (continued)

~~The 7 day Completion Time is reasonable, based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable, the number of available systems, and the time required to complete the action.]~~ 4

A A

~~B~~.1 and ~~B~~.2 1

Insert 2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

~~If the cooling tower fan cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours.~~

~~Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 3). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.~~

~~Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and~~

1

INSERT 2

With the UHS inoperable, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to MODE 3 and the backup cooling water supply must be aligned to the intake canal. Risk in MODE 3 is minimized by providing makeup from a backup water source to the intake canal. This action ensures the backup supply is aligned to restore cooling water for the ICW pumps if the primary supply cannot be restored to within limit. Also, there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms in MODES 3 and 4 than in MODE 5. Voluntary entry into MODE 4 or 5 may be made as these MODES are also acceptable low-risk states. Required Action A.2 is provided to supply cooling water from the backup water supply within 18 hours. The Completion Time for Required Action A.2 allows the operator time to evaluate and repair any discovered inoperabilities while providing adequate time to align the backup supply to the intake canal. If the backup water supply cannot be established within the required Completion Time, Condition B must be entered.

BASES

ACTIONS (continued)

establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power in an orderly manner and without challenging plant systems.

[C.1

REVIEWER'S NOTE

The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit.

With water temperature of the UHS > [90]°F, the design basis assumption associated with initial UHS temperature are bounded provided the temperature of the UHS averaged over the previous 24 hour period is ≤ [90]°F. With the water temperature of the UHS > [90]°F, long term cooling capability of the ECCS loads and DGs may be affected. Therefore, to ensure long term cooling capability is provided to the ECCS loads when water temperature of the UHS is > [90]°F, Required Action C.1 is provided to more frequently monitor the water temperature of the UHS and verify the temperature is ≤ [90]°F when averaged over the previous 24 hour period. The once per hour Completion Time takes into consideration UHS temperature variations and the increased monitoring frequency needed to ensure design basis assumptions and equipment limitations are not exceeded in this condition. If the water temperature of the UHS exceeds [90]°F when averaged over the previous 24 hour period or the water temperature of the UHS exceeds [ ]°F, Condition D must be entered immediately.] 5

B

[D.1 and D.2 3

A are

If the Required Actions or Completion Times of Condition C is not met, or the UHS is inoperable [for reasons other than Condition A or C], the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.] 2 1

is

BASES

SURVEILLANCE REQUIREMENTS

~~[~~ SR 3.7.9.1

This SR verifies adequate long term (30 days) cooling can be maintained. The level specified also ensures sufficient NPSH is available for operating the ~~SWS~~ pumps. ~~[The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ [562] ft [mean sea level].~~

ICW

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]]~~

2

~~[SR 3.7.9.2~~

~~This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature within the maximum accident or normal design heat loads for 30 days following a DBA. [The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water temperature is ≤ [92]°F.~~

5

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.7.9.3

Operating each cooling tower fan for ≥ [15] minutes verifies that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration can be detected for corrective action. [The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances. 4

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]]

U

REFERENCES

1. FSAR, Section [9.2.5]. 1 2

2. Regulatory Guide 1.27.

3. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001. 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.9, BASES, ULTIMATE HEAT SINK (UHS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS ACTIONS numbering.

4. The PSL UHS design does not include cooling towers, therefore, ISTS 3.7.9 Bases ACTION A.1 discussion and ISTS SR 3.7.9.3 discussion are not included in the ITS Bases.

5. ITS 3.7.9 does not include UHS water temperature, therefore, ISTS 3.7.9 Bases ACTION C and ISTS SR 3.7.9.2 are not included in the ITS Bases.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.9, ULTIMATE HEAT SINK (UHS)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 10

ITS 3.7.10, Control Room Emergency Ventilation System (CREVS)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

PLANT SYSTEMS

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

(CREVS)

LIMITING CONDITION FOR OPERATION

A02

LCO 3.7.10

3.7.7.1 Two CREVS trains ~~The control room emergency ventilation system~~ shall be OPERABLE ~~with:~~

~~a. Two booster fans,~~

~~b. Two isolation valves in each outside air intake duct,~~

~~c. Two isolation valves in the toilet area air exhaust duct,~~

~~d. One filter train,~~

e. At least two air conditioning units, and

See ITS 3.7.11

~~f. Two isolation valves in the kitchen area exhaust duct~~.

LA01

LCO 3.7.10 NOTE

NOTE

The control room envelope boundary may be opened intermittently under administrative control.

Applicability

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6 or during movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3 and 4:

for reasons other than Condition B

One CREVS train

CREVS train

ACTION A

a. ~~With one booster fan~~ inoperable, restore ~~the inoperable fan~~ to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within ~~the following 6~~ hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION F

Required Action F.2 NOTE

12

b. ~~With one isolation valve per air duct inoperable, operation may continue provided the other isolation valve in the same duct is maintained closed; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN~~.

L01

Add proposed NOTE to ACTION C

M01

Two CREVS trains

A02

c. ~~With the filter train~~ inoperable for reasons other than an inoperable Control Room Envelope boundary:

ACTION C

1. Immediately initiate action to implement mitigating actions, and
2. Within 1 hour, verify LCO 3.4.~~8~~, "Specific Activity," is met, and

16

3. Within 24 hours restore the filter train to OPERABLE status.

ACTION F

MODE 4

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and ~~COLD SHUTDOWN within the following 30~~ hours.

12

L02

d. With only one air conditioning unit OPERABLE, restore at least two air conditioning units to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.7.11

PLANT SYSTEMS

ACTION: (continued)

MODES 1, 2, 3 and 4: (continued)

ACTION C NOTE

NOTE

Action not applicable when second ~~booster fan~~ intentionally made inoperable.

CREVS train

ACTION C

e. ~~With~~ two ~~booster fans~~ inoperable for reasons other than ~~an inoperable Control Room Envelope boundary~~:

CREVS trains

Condition B

1. Immediately initiate action to implement mitigating actions, and

2. Within 1 hour, verify LCO 3.4.~~8~~16, "Specific Activity," is met, and

3. Within 24 hours restore at least one ~~booster fan~~ CREVS train to OPERABLE status.

ACTION F

MODE 4

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and in ~~COLD SHUTDOWN~~ within ~~the following~~ 30 hours. 12

L02

Required Action F.2 NOTE

Add proposed Required Action F.2 NOTE

NOTE

Action not applicable when third air conditioning unit intentionally made inoperable.

f. With three air conditioning units inoperable for reasons other than an inoperable Control Room Envelope boundary, restore at least one air conditioning unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

g. ~~With the filter train~~ inoperable due to an inoperable Control Room Envelope boundary:

One or more CREVS trains

See ITS 3.7.11

ACTION B

1. Immediately initiate actions to implement mitigating actions, and

2. Within 24 hours, verify mitigating actions to ensure Control Room Envelope occupant exposures to radiological, chemical, and smoke hazards will not exceed limits, and

3. Restore Control Room Envelope boundary to OPERABLE status within 90 days.

ACTION F

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within ~~the following~~ 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

12

Required Action F.2 NOTE

PLANT SYSTEMS

ACTION: (continued)

MODES 5 and 6 or during movement of irradiated fuel assemblies:

ACTION A

a. ~~With one booster fan~~ inoperable, restore the inoperable fan to OPERABLE status within 7 days or ~~initiate and maintain operation of the remaining~~ OPERABLE ~~control room emergency ventilation system~~ in the recirculation mode or suspend movement of irradiated fuel assemblies.

ACTION D

One CREVS train

for reasons other than Condition B

place

CREVS train

b. ~~With one isolation valve in an air duct inoperable, maintain the other isolation valve in the same air duct closed or suspend movement of irradiated fuel assemblies.~~

L01

Two CREVS trains inoperable or with one or more CREVS trains inoperable due to an inoperable CRE boundary, immediately

ACTION E

c. ~~With the filter train inoperable,~~ suspend movement of irradiated fuel assemblies.

See ITS 3.7.11

d. With only one air conditioning unit OPERABLE, restore at least two air conditioning units to OPERABLE status within 7 days or suspend movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.7.1 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is ≤ 120°F.

b. In accordance with the Surveillance Frequency Control Program by:

See ITS 3.7.11

SR 3.7.10.1

1. ~~Initiating flow through the HEPA filter and charcoal adsorber train and verifying that each booster fan~~ operate~~s for at least 15~~ minutes.

LA02

each CREV train for ≥ 15 continuous

See ITS 3.7.11

2. Starting (unless already operating) each air conditioning unit and verifying that it operates for at least 8 hours.

CREVS

SR 3.7.10.2

c. ~~By~~ performing required ~~control room emergency ventilation system~~ filter testing in accordance with the Ventilation Filter Testing Program.

~~**PLANT SYSTEMS**~~

~~**SURVEILLANCE REQUIREMENTS** (Continued)~~

~~DELETED~~

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.7.10.3

d. In accordance with the Surveillance Frequency Control Program by verify~~ing that on a containment isolation signal the system automatically isolates the control room within 35 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks~~.

L03 L04 LA03

SR 3.7.10.4

e. ~~By~~ p~~erform~~ing required Control Room Envelope unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

each CREVS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position and isolates the control room in ≤ 35 seconds.

PLANT SYSTEMS

VENTILATION

CREVS

3/4.7.7 CONTROL ROOM EMERGENCY ~~AIR CLEANUP~~ SYSTEM (~~CREACS~~)

LIMITING CONDITION FOR OPERATION

A02

LCO 3.7.10

3.7.7 Two ~~independent control room emergency air cleanup systems~~ shall be OPERABLE ~~with:~~

CREVS trains

~~a. A filter train and its associated fan per system, and~~

b. At least one air conditioning unit per system, and

See ITS 3.7.11

~~c. Two isolation valves in the kitchen area exhaust duct, and~~

~~d. Two isolation valves in the toilet area exhaust duct, and~~

~~e. Two isolation valves in each (North and South) air intake duct~~.

LA01

LCO 3.7.10 NOTE

NOTE

The control room envelope boundary may be opened intermittently under administrative control.

Applicability

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6 or during movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

One CREVS train

a. ~~With one control room emergency air cleanup system~~ inoperable for reasons other than ~~an inoperable Control Room Envelope boundary~~, restore ~~the inoperable system~~ to OPERABLE status within 7 days or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within ~~the following~~ 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Condition B

ACTION A

CREVS train

ACTION F

Required Action F.2 NOTE

12

CREVS trains

b. ~~With~~ one or more ~~control room emergency air cleanup systems~~ inoperable due to an inoperable Control Room Envelope boundary:

ACTION B

1. Immediately initiate actions to implement mitigating actions, and
2. Within 24 hours, verify mitigating actions to ensure Control Room Envelope occupant exposures to radiological, chemical, and smoke hazards will not exceed limits, and
3. Restore Control Room Envelope boundary to OPERABLE status within 90 days.

12

ACTION F

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within ~~the following~~ 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Required Action F.2 NOTE

c. ~~With an isolation valve in an air intake duct or air exhaust duct inoperable, operation may continue provided the other isolation valve in the same air intake or air exhaust duct is maintained closed; otherwise be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.~~

L01

ACTION C NOTE

CREVS

NOTE

Action not applicable when second ~~CREACS~~ train intentionally made inoperable.

Two CREVS trains

d. ~~With two control room emergency air cleanup systems~~ inoperable for reasons other than ~~an inoperable Control Room Envelope boundary~~.

Condition B

ACTION C

1. Immediately initiate actions to implement mitigating actions, and
2. Within 1 hour, verify LCO 3.4.~~8~~, "Specific Activity," is met, and

16

CREVS train

3. Within 24 hours restore at least one ~~CREACS~~ train to OPERABLE status.

ACTION F

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and ~~COLD SHUTDOWN~~ within ~~the following~~ 30 hours.

MODE 4

12

L02

Add proposed Required Action F.2 NOTE

PLANT SYSTEMS

ACTION: (continued)

MODES 5 and 6 or during movement of irradiated fuel assemblies:

ACTION A

a. ~~With one control room emergency air cleanup system~~ inoperable for reasons other than ~~an inoperable Control Room Envelope boundary~~, immediately ~~initiate and maintain operation of the remaining~~ OPERABLE ~~control room emergency air cleanup system~~ in the recirculation mode or immediately suspend movement of irradiated fuel assemblies.

ACTION D

One CREVS train

Condition B

CREVS train

place

ACTION E

b. ~~With both control room emergency air cleanup systems~~ inoperable, or with one or more ~~CREACS systems~~ inoperable due to an inoperable Control Room Envelope boundary, immediately suspend movement of irradiated fuel assemblies.

Two CREVS trains

CREVS trains

c. ~~With an isolation valve in an air intake duct or air exhaust duct inoperable, maintain the other isolation valve in the same air intake or air exhaust duct closed or suspend movement of irradiated fuel assemblies.~~

L01

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

See ITS 3.7.11

4.7.7 Each control room emergency air cleanup system shall be demonstrated OPERABLE~~:~~

a. In accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is ≤ 120°F.

each CREV train for ≥ 15 continuous

SR 3.7.10.1

b. In accordance with the Surveillance Frequency Control Program by (1) ~~initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system~~ operate~~s for at least 15~~ minutes ~~and~~ (2) starting, unless already operating each air conditioning unit and verifying that it operates for at least 8 hours.

LA02

CREVS

SR 3.7.10.2

c. ~~By~~ perform~~ing~~ required ~~control room emergency air cleanup system~~ filter testing in accordance with the Ventilation Filter Testing Program.

See ITS 3.7.11

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

d. In accordance with the Surveillance Frequency Control Program by:

1. Verify~~ing that on a containment isolation test signal from Unit 2, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.~~

SR 3.7.10.3

each CREVS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.

L03 L04 LA03

2. ~~Verifying that on a containment isolation test signal from Unit 1 the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.~~

SR 3.7.10.4

e. ~~By~~ perform~~ing~~ required Control Room Envelope unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

DISCUSSION OF CHANGES
ITS 3.7.10, CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.7.7.1 and Unit 2 CTS 3.7.7 provide requirements for the Control Room Emergency Ventilation System (CREVS) and include requirements for control room air conditioning. ITS 3.7.10, “Control Room Emergency Ventilation System (CREVS),” provides requirements for the control room filtration function and ITS 3.7.11, “Control Room Air Conditioning System (CRACS),” provides requirements for the control room cooling function. This changes the CTS by separating the CREVS and CRACS requirements into separate Specifications based on function.

Although CREVS and CRACS are subsystems within the Control Room Ventilation System and utilize some of the same components and ductwork, these systems provide two separate and distinct safety functions. Therefore, the ITS presentation allows the filtration and cooling functions to be addressed separately. This presentation change is consistent with the ISTS and is designated as administrative because it does not result in technical changes to the CTS. Any technical changes associated with the current CREVS and CRACS requirements are provided in the Discussion of Changes (DOCs) associated with that Specification.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 Only**: CTS 3.7.7.1 Action c (MODES 1, 2, 3, and 4) provide remedial actions when the filter train is inoperable for reasons other than an inoperable control room envelop boundary, including a time to restore the filter train within 24 hours. ITS 3.7.10 ACTION C requires similar actions when two CREVS trains are inoperable (e.g., the common filter train inoperable) in MODE 1, 2, 3, or 4 for reasons other than Condition B (i.e., inoperable control room envelop boundary) and includes a Note that does not allow these actions when a second CREVS train is intentionally made inoperable. This changes the CTS related to an inoperable filter train to only allow the actions when the filter train is discovered inoperable and not allow these actions if the single filter train is removed from service intentionally.

The purpose of the CTS actions is to allow a short period of time to restore the CREVS to OPERABLE status during the condition with no OPERABLE control room filter trains. However, this allowance is intended to only be used if the loss of filter function was unintentional. Voluntary removal of the filter safety function is not allowed, as indicated in the Note to CTS 3.7.7.1 Action e. The PSL Unit 1

CREVS design includes two redundant trains of active equipment and a single filter unit that is common to both CREVS trains. Therefore, when the filter unit is inoperable, both CREVS trains are inoperable and the control room filter safety function cannot be performed. Therefore, the CTS 3.7.7.1 Action c and Action e are combined in ITS 3.7.10 ACTION C and include the Note that does not allow these actions when a second CREVS train is intentionally made inoperable. This change will preclude the common filter unit from intentionally being removed from service that would result in a voluntarily removal of a required safety function. This change is designated as more restrictive because it adds a restriction that CTS does not have.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.7.7.1 states that the control room ventilation system shall be OPERABLE with; two booster fans, two isolation valves in each outside air intake duct, two isolation valves in the toilet area exhaust duct, one filter train, at least two air conditioning units and two isolation valves in the kitchen area exhaust duct. Unit 2 CTS 3.7.7 states that two independent control room emergency air cleanup systems shall be OPERABLE with; a filter train and its associated fan per system, at least one air conditioning unit per system, two isolation valves in the kitchen area exhaust duct, two isolation valves in the toilet area exhaust duct, and two isolation valves in each (North and South) air intake duct. ITS LCO 3.7.10 states that two CREV trains shall be OPERABLE, but the details of what constitutes an OPERABLE CREV train are moved to the Bases. This changes the CTS by removing details of what constitutes an OPERABLE train to the Bases.

The removal of these details, which are related to system design, from CTS 3.7.7 is acceptable because the information is not necessary within the technical specification to provide adequate protection of public health and safety. This change is acceptable because ITS retains the requirement for two CREVS trains to be OPERABLE and retains the detail of what constitutes a CREVS train within the Bases. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 2 – Removing Descriptions of System Operation)* Unit 1 CTS 4.7.7.1.b.1 states, in part, “initiating flow through the HEPA filter and charcoal adsorber train and verifying that each booster fan.” Unit 2 CTS 4.7.7.b states, in part, “initiating, from the control room, flow through the HEPA filters and charcoal adsorbers.” ITS SR 3.7.10.1 states, “Operate each CREVS train for ≥ 15 continuous minutes.” This changes the CTS by moving the system operational detail to the UFSAR.

The removal of these details, which are related to system operation, from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change, because information relating to system operation is being removed from the Technical Specifications.

LA03 *(Type 2 – Removing Descriptions of System Operation)* Unit 1 CTS 4.7.7.1.d states, in part, "that on a containment isolation signal the system automatically isolates the control room and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks." Unit 2 CTS 4.7.7.d.1 and CTS 4.7.7.d.2 state, in part, "that on a containment isolation test signal the system automatically isolates the control room and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks." ITS SR 3.7.10.3 states, "Verify each CREVS train actuates on an actual or simulated actuation signal." This changes the CTS by moving the system operational detail to the UFSAR. (See DOCs L01 and L02 for additional discussion.)

The removal of these details, which are related to system operation, from the Technical Specifications is acceptable, because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change, because information relating to system operation is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.7.7.1 Actions b (MODES 1, 2, 3, and 4) and b (MODES 5 and 6 or during movement of irradiated fuel assemblies) and Unit 2 CTS 3.7.7 Actions c (MODES 1, 2, 3, and 4) and c (MODES 5 and 6 or during movement of irradiated fuel assemblies) provide actions when an isolation valve in an air intake duct or an air exhaust duct is inoperable. ITS 3.7.10 ACTIONS do not include specific actions for an inoperable valve in an air intake duct or an air exhaust duct. This changes the CTS by removing specific action requirements associated with an inoperable isolation valve in a CREVS intake or exhaust duct.

The purpose of the CTS actions is to ensure the control room envelop boundary is maintained when redundant isolation capability is lost on a CREVS intake or exhaust duct. The CREVS design includes dual valves on the common intake and exhaust ducts to ensure these penetrations are isolated in the event of accident requiring control room isolation. The series valves on each air intake and air exhaust duct are powered and actuated from redundant trains. Therefore, when one isolation valve in an air duct is inoperable, appropriate

action for that level of degradation is to declare the CREV train associated with the inoperable isolation valve inoperable. Additionally, closing and deactivating either valve in the isolated position ensures the safety function is performed and restores the LCO. Unit 1 CTS 4.7.7.1.d and Unit 2 CTS 4.7.7.d.1 and 2 requires CREV components to actuate on a containment isolation signal, which includes automatic isolation valves. ITS SR 3.7.10.3 also requires similar testing with an exception for dampers and valves that are locked, sealed, or otherwise secured in the actuated position (See DOC L04 for the addition of this exception). Therefore, the exception in the SR effectively allows continued operation when one or more isolation valve in an air intake or exhaust duct are inoperable, provided the isolation valves are locked, sealed, or otherwise secured in the actuated (i.e., isolated) position. This change is acceptable because actions retained in the ITS continue to ensure redundant isolation capability is restored or the isolation function is performed. These remedial actions, including the exception provided in ITS SR 3.7.10.3, are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant isolation valve and the low probability of a DBA occurring during this condition. This change is designated as less restrictive because removal of the CTS actions allows restoration of the isolation capability to be controlled by other ITS 3.7.10 actions, which are less restrictive and allows more than one isolation valve per air duct to be inoperable provided the isolation function is performed.

L02 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.7.7.1 Actions c and e (MODES 1, 2, 3 and 4) require when the action requirements for a filter train inoperability for reasons other than an inoperable control room envelope boundary or for the inoperability of two booster fans, respectively, cannot be met the unit must be in at least HOT STANDBY (MODE 3) with 6 hours and COLD SHUTDOWN (MODE 5) within the following 30 hours. Unit 1 CREVS trains share a common filter train, therefore, CTS Actions c and e are effectively addressing the same condition and are combined in ITS as ACTION E. Unit 2 CTS 3.7.7 Action d (MODES 1, 2, 3 and 4) requires when two control room emergency air cleanup systems are inoperable for reasons other than an inoperable control room envelope boundary and the specified required actions and completion times cannot be met that the unit be in at least HOT STANDBY (MODE 3) with 6 hours and COLD SHUTDOWN (MODE 5) within the following 30 hours. ITS ACTION E requires for a similar condition, to be in MODE 3 within 6 hours and in MODE 4 within 12 hours with a Note stating that LCO 3.0.4.a is not applicable when entering MODE 4. This changes the CTS by changing the end state from “COLD SHUTDOWN within the following 30 hours” to “MODE 4 within 12 hours with LCO 3.0.4.a allowance not applicable.”

The purpose in the CTS action to shutdown the unit when the LCO cannot be restored within the required Completion Time is to place the unit in a MODE that minimizes overall plant risk. Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 as indicated in CE-NPSD-1186-A, Revision 00, “Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs,” dated October 2001 (ADAMS Accession No. 110410539). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. Voluntary entry into MODE 5 may continue to be made as it is also an acceptable

low-risk state. The change is consistent with similar changes made to CTS 3.7.7 in Amendments 234 and 184, “St. Lucie Plant, Unit Nos. 1 and 2 – Issuance of Amendments Regarding Changes in Selected Technical Specifications End States (CAC Nos. MF8106 and MF8107; Formerly CAC NOS. MF6683 AND MF6684),” dated August 30, 2016, (ADAMS Accession No. ML16210A374). This change is acceptable because the modification of the end state from MODE 5 to MODE 4 is consistent with CE-NPSD-1186-A and corrects an inconsistency in the CTS end states for Actions intended to place the unit in a condition outside of the LCO Applicability. Therefore, this change corrects that oversight. This change is designated as less restrictive because instead of requiring the plant to achieve MODE 5 (COLD SHUTDOWN), the ITS end state is MODE 4 (HOT SHUTDOWN) with LCO 3.0.4.a allowance not applicable.

L03 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.7.7.1.d requires verification that on a containment isolation signal the system automatically isolates the control room and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.” Unit 2 CTS 4.7.7.d.1 and d.2 state, in part, “that on a containment isolation test signal the system automatically isolates the control room and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.” ITS SR 3.7.10.3 specifies that the signal may be either an "actual" or a simulated actuation signal. This changes the Unit 1 and Unit 2 CTS by allowing the use of either an actual or a simulated signal for the test.

The purpose of CTS Surveillances is to ensure that each CREVS train actuates (switches to the recirculation mode) upon receipt of an actuation signal. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual," "simulated," or "test" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L04 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.7.7.1.d requires verification that on a containment isolation signal the system automatically isolates the control room and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.” Unit 2 CTS 4.7.7 d.1 and d.2 require verification that Unit 1 and Unit 2 containment isolation test signals will automatically switch into a recirculation mode of operation. ITS SR 3.7.10.3 requires verification that each CREVS train actuates and additional states, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position. This changes the CTS by not requiring dampers and valves locked, sealed or otherwise secured in position to be tested to verify that they automatically actuate to their actuated position.

The purpose of the CTS Surveillances is to provide assurance that the dampers and valves required to actuate in case of a design basis accident (DBA) isolate the control room and shift CREVS into the recirculation mode. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Dampers and valves already in the isolated position and secured are not required to be tested to automatically actuate because, in case of a DBA, they are already in their required position and secured to prevent changing from the required position. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

ITS

3.7 PLANT SYSTEMS

3.7.11 Control Room Emergency Air Cleanup System (CREACS)

3.7.7.1 DOC LA01

LCO 3.7.11 Two CREACS trains shall be OPERABLE.

LCO 3.7.7.1 NOTE

---NOTE---
The control room envelope (CRE) boundary may be opened intermittently under administrative control.

Applicability

APPLICABILITY: MODES 1, 2, 3, 4, [5, and 6,]
During movement of [recently] irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a A. One CREACS train inoperable for reasons other than Condition B.	A.1 Restore CREACS train to OPERABLE status.	7 days
Action g B. One or more CREACS trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4.	B.1 Initiate action to implement mitigating actions. AND B.2 Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits. AND B.3 Restore CRE boundary to OPERABLE status.	Immediately 24 hours 90 days

ITS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action e NOTE C. ---------- NOTE ---------- Not applicable when second CREACS train intentionally made inoperable. ---------- Action e Two CREACS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 Initiate action to implement mitigating actions. AND C.2 Verify LCO 3.4.16, "RCS Specific Activity," is met. AND C.3 Restore at least one CREACS train to OPERABLE status.	Immediately 1 hour 24 hours
Action a D. Required Action and associated Completion Time of Condition A not met ~~[~~in MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies.	D.1 ~~---------- NOTE ----------~~ ~~Place in toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable.~~ ~~----------~~ Place OPERABLE CREACS train in ~~emergency radiation protection~~ mode. (recirculation) OR D.2 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately Immediately

ITS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action c E. Two CREACS trains inoperable [in MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies. OR One or more CREASC trains inoperable due to an inoperable CRE boundary [in MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies.	E.1 Suspend movement of [recently] irradiated fuel assemblies.	Immediately
Action a, b, c, e and g F. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, 3, or 4.	F.1 Be in MODE 3. AND F.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

Action a, b c, e and g DOC L01

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.7.7.1.b	SR 3.7.11.1	Operate each CREACS train for ≥ 15 continuous minutes [with heaters operating].	[31 days OR In accordance with the Surveillance Frequency Control Program]
4.7.7.1.c	SR 3.7.11.2	Perform required CREACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]
4.7.7.d.1 and d.2 DOC L02 DOC L03	SR 3.7.11.3	Verify each CREACS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	[[18] months OR In accordance with the Surveillance Frequency Control Program]
4.7.7.e	SR 3.7.11.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

ITS

3.7 PLANT SYSTEMS

3.7.11 Control Room Emergency Air Cleanup System (CREACS)

3.7.7 DOC LA01

LCO 3.7.11 Two CREACS trains shall be OPERABLE.

LCO 3.7.7 NOTE

---NOTE---
The control room envelope (CRE) boundary may be opened intermittently under administrative control.

Applicability

APPLICABILITY: MODES 1, 2, 3, 4, [5, and 6,]
During movement of [recently] irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a A. One CREACS train inoperable for reasons other than Condition B.	A.1 Restore CREACS train to OPERABLE status.	7 days
Action b B. One or more CREACS trains inoperable due to inoperable CRE boundary in MODE 1, 2, 3, or 4.	B.1 Initiate action to implement mitigating actions. AND B.2 Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits. AND B.3 Restore CRE boundary to OPERABLE status.	Immediately 24 hours 90 days

ITS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action d NOTE C. ---------- NOTE ---------- V Not applicable when second CREACS train intentionally made inoperable. ---------- Action d V Two CREACS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 Initiate action to implement mitigating actions. AND C.2 Verify LCO 3.4.16, "RCS Specific Activity," is met. AND C.3 V Restore at least one CREACS train to OPERABLE status.	Immediately 1 hour 24 hours
Action a D. Required Action and associated Completion Time of Condition A not met ~~[~~in MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies.	D.1 ~~---------- NOTE ---------- Place in toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable. ----------~~ V Place OPERABLE CREACS train in ~~emergency radiation protection~~ mode. recirculation OR D.2 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately Immediately

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ITS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b E. Two CREACS trains inoperable [in MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies. OR One or more CREASC trains inoperable due to an inoperable CRE boundary [in MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies.	E.1 Suspend movement of [recently] irradiated fuel assemblies.	Immediately
Action a, b, c and d F. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, 3, or 4.	F.1 Be in MODE 3. AND F.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

Action a, b and c DOC L01

ITS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.7.7.b	SR 3.7.11.1	Operate each CREACS train for ≥ 15 continuous minutes [with heaters operating].	[31 days OR In accordance with the Surveillance Frequency Control Program]
4.7.7.c	SR 3.7.11.2	Perform required CREACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]
4.7.7.d.1 and d.2 DOC L02 DOC L03	SR 3.7.11.3	Verify each CREACS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	[[18] months OR In accordance with the Surveillance Frequency Control Program]
4.7.7.e	SR 3.7.11.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.10, CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS 3.7.11 title "Control Room Emergency Air Cleanup System (CREACS)" has been changed to "Control Room Emergency Ventilation System (CREVS)" consistent with the St. Lucie Plant (PSL) site specific terminology. In addition, PSL design does not include the Essential Chilled Water System (ISTS 3.7.10). Therefore, ISTS 3.7.10 is not included in the PSL ITS. The Control Room Emergency Air Cleanup System (ISTS 3.7.11) is renumbered as ITS 3.7.10.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

10
Ventilation
CREVS
B 3.7.~~11~~ Control Room Emergency ~~Air Cleanup~~ System (~~CREACS~~)
2
1

BASES

BACKGROUND

CREVS
The ~~CREACS~~ provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity~~, hazardous chemicals, or smoke~~.

CREVS
The ~~CREACS~~ consists of two ~~independent,~~ redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each ~~CREACS~~ train consists of a prefilter ~~and demister~~, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system~~, as well as demisters that remove water droplets from the air stream~~. ~~A second bank of HEPA filters follows the adsorber section to collect carbon fines, and provides back up in case of failure of the main HEPA filter bank.~~

CREVS
f
shares a common filter train that
Also included within a filter train are one booster fan, at least one air conditioning unit, two kitchen area exhaust duct isolation valves, two toilet area exhaust duct isolation valves and two air intake duct isolation valves in each (north and south) intake duct.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, ~~and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident~~. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of Design Basis Accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

kitchen and food storage room, restrooms, administrative offices, computer room, and technical support center.

1

CREVS
(ITS 5.5.15)
s
The ~~CREACS~~ is an emergency system, part of which ~~may~~ also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the CRE is isolated, and the stream of ventilation air is recirculated through the filter trains of the system. The prefilters ~~and demisters~~ remove any large particles in the air~~, and any entrained water droplets present~~ to prevent excessive loading of the HEPA filters and charcoal adsorbers. ~~Both the demister and heater are important to the effectiveness of the charcoal adsorbers.~~

BASES

BACKGROUND (continued)

CREVS

Actuation of the ~~CREACS~~ places the system into ~~either of two separate states of the emergency mode of operation, depending on the initiation signal. Actuation of the system to the emergency radiation state of the emergency~~ mode of operation closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the ~~redundant trains of~~ HEPA and charcoal ~~filters~~. The ~~emergency radiation state also initiates~~ pressurization and filtered ventilation of the air supply to the CRE.

the recirculation
which
starts control room emergency cleanup fans, standby control room air conditioning unit,
adsorber
filter
recirculation mode also facilitates
1

Outside air is filtered~~, [diluted with building air from the electrical equipment and cable spreading rooms,]~~ and then added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. ~~The actions taken in the toxic gas isolation state are the same, except that the signal switches the CREACS to an isolation mode, minimizing outside air entering the CRE through the CRE boundary.~~

3

The air entering the CRE is continuously monitored by radiation ~~and toxic gas detectors~~. One detector output above the setpoint will cause actuation of the ~~emergency radiation state or toxic gas isolation state~~ as required. ~~The actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.~~

monitors
recirculation mode

In recirculation mode, a makeup
450
is acceptable to
surrounding
CREVS
U
and 6.4 (Ref. 3)

~~A single CREACS train operating at a~~ flow rate of ≤ ~~[~~3000~~]~~ cfm ~~will~~ pressurize the CRE to ~~about [~~0.125~~]~~ inches water gauge relative to ~~external~~ areas adjacent to the CRE boundary~~, and provides an air exchange rate in excess of 25% per hour~~. The ~~CREACS~~ operation in maintaining the CRE habitable is discussed in the FSAR, Sections ~~[~~9.4~~]~~ (Ref. 1).

3
1
3

~~Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train~~. Normally open isolation ~~dampers~~ are arranged in series pairs so that the failure of one ~~damper~~ to shut will not result in a breach of isolation. The ~~CREACS~~ is designed in accordance with Seismic Category I requirements.

valves
valve
CREVS

CREVS

The ~~CREACS~~ is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a DBA without exceeding a ~~[5 rem whole body dose or its equivalent to any part of the body]~~ [5 rem total effective dose equivalent (TEDE)~~]~~.

3

BASES

APPLICABLE SAFETY ANALYSES

CREVS

The ~~CREACS~~ components are arranged in redundant safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access.

which share a common filter train

CREVS

The ~~CREACS~~ provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident fission product release presented in the FSAR, Chapter ~~[~~15~~]~~ (Ref. 2).

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U

3

CREVS

The ~~CREACS~~ provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 3). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels ~~(Ref. 4)~~.

CREVS

The worst case single active failure of a component of the ~~CREACS~~, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

1

CREVS

The ~~CREACS~~ satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

CREVS

Two ~~independent and~~ redundant trains of the ~~CREACS~~ are required to be OPERABLE to ensure that at least one is available, if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of ~~[5 rem whole body or its equivalent to any part of the body]~~ [5 rem TEDE~~]~~ ~~or~~ the CRE occupants in the event of a large radioactive release.

1

3

to

CREVS

Each ~~CREACS~~ train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE.

CREVS

A ~~CREACS~~ train is considered OPERABLE when the associated:

a. Fan is OPERABLE,

b. HEPA filters and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions, and

c. ~~Heater, demister,~~ ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

1

BASES

LCO (continued)

CREVS
In order for the ~~CREACS~~ trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke. 1

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, 4, ~~[~~5, and 6~~]~~, and during movement of ~~[recently]~~ irradiated fuel assemblies, the ~~CREACS~~ must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA. 3
CREVS

~~In MODES [5 and 6], the CREACS is required to cope with the release from a rupture of an outside waste gas tank.~~ 1

CREVS
During movement of ~~[recently]~~ irradiated fuel assemblies, the ~~CREACS~~ must be OPERABLE to cope with the release from a fuel handling accident ~~[involving handling recently irradiated fuel]~~. ~~[Due to radioactive decay, the CREACS is only required to cope with fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days).]~~ 3 3

ACTIONS

A.1

CREVS
With one ~~CREACS~~ train inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE ~~CREACS~~ train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE ~~CREACS~~ train could result in loss of ~~CREACS~~ function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability. 1
CREVS
CREVS

BASES

ACTIONS (continued)

B.1, B.2, and B.3

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to ~~[5 rem whole body or its equivalent to any part of the body]~~ [5 rem TEDE~~]~~), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days. 3

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

C.1, C.2, and C.3

CREVS
If both ~~CREACS~~ trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable control room boundary (i.e., Condition B), at least one ~~CREACS~~ train must be returned to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second ~~CREACS~~ train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be

CREVS
CREVS
1

BASES

ACTIONS (continued)

CREVS inoperable, or if both trains are discovered to be inoperable at the same time. During the period that the ~~CREACS~~ trains are inoperable, action must be initiated to implement mitigating actions to lessen the effect on CREVS CRE occupants from potential hazards while both trains of ~~CREACS~~ are inoperable. In the event of a DBA, the mitigating actions will reduce the consequences of radiological exposures to the CRE occupants.

Specification 3.4.16, "RCS Specific Activity," allows limited operation with the reactor coolant system (RCS) activity significantly greater than the LCO limit. This presents a risk to the plant operator during an accident CREVS when all ~~CREACS~~ trains are inoperable. Therefore, it must be verified within 1 hour that LCO 3.4.16 is met. This Required Action does not require additional RCS sampling beyond that normally required by LCO 3.4.16. 1

CREVS At least one ~~CREACS~~ train must be returned to OPERABLE status within 24 hours. The Completion Time is based on Reference 3 which demonstrated that the 24 hour Completion Time is acceptable based on the infrequent use of the Required Actions and the small incremental effect on plant risk.

D.1 and D.2

~~[~~In MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies, if Required Action A.1 cannot be completed within the CREVS required Completion Time, the OPERABLE ~~CREACS~~ train must be recirculation immediately placed in the ~~emergency~~ mode of operation. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected. 3 1

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position.

~~Required Action D.1 is modified by a Note indicating to place the system in the toxic gas protection mode if the automatic transfer to the toxic gas protection mode is inoperable.~~ 1

BASES

ACTIONS (continued)

E.1

When ~~[~~in MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies, with two ~~CREACS~~ trains inoperable or with one or more ~~CREACS~~ trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

CREVS
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F.1 and F.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

4

CREVS

If the inoperable ~~CREACS~~ or control room boundary cannot be restored to OPERABLE status within the associated Completion Time in MODE 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~5~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

4

1

Required Action F.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable,

BASES

ACTIONS (continued)

because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~11~~.1 (10) 5

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Operation ~~[with the heaters on]~~ for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that ~~[heater failure,]~~ blockage, fan or motor failure, or excessive vibration can be detected for corrective action. ~~[The 31 day Frequency is based on the known reliability of the equipment, and the two train redundancy available.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

10
2
SR 3.7.~~11~~.2

CREVS
1
This SR verifies that the required ~~CREACS~~ testing is performed in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. The ~~[~~VFTP~~]~~ includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~.
3

(ITS 5.5.8)

10
2
SR 3.7.~~11~~.3

CREVS
1
This SR verifies that each ~~CREACS~~ train starts and operates on an actual or simulated actuation signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. ~~[The Frequency of [18] months is based on industry operating experience and is consistent with the typical refueling cycle.~~
3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
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BASES

SURVEILLANCE REQUIREMENTS (continued)

(10) SR 3.7.~~11~~.4 (2)

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than ~~[5 rem whole body or its equivalent to any part of the body]~~ [5 rem TEDE~~]~~ and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. ~~6~~) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. ~~7~~). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. ~~8~~). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

(3) (5) (6) (4) (7)

REFERENCES

1. FSAR, Section ~~[~~9.4~~]~~. (U)
2. FSAR, Chapter ~~[~~15~~]~~. (U)
3. FSAR, Section ~~[~~6.4~~]~~. (U)

(1) (3)

~~4. FSAR, Section [9.5].~~

(4) ~~5~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

(5) ~~6~~. Regulatory Guide 1.196.

(1)

BASES

REFERENCES (continued)

6 ~~7~~. NEI 99-03, "Control Room Habitability Assessment," June 2001.

7 ~~8~~. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040300694).

8 ~~9~~. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

1

B 3.7 PLANT SYSTEMS

10
B 3.7.~~11~~ Control Room Emergency ~~Air Cleanup~~ System (~~CREACS~~)
Ventilation
CREVS
2
1

BASES

BACKGROUND

The ~~CREACS~~ provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity~~, hazardous chemicals, or smoke~~.

CREVS

CREVS

The ~~CREACS~~ consists of two ~~independent,~~ redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each ~~CREACS~~ train consists of a prefilter ~~and demister~~, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system~~, as well as demisters that remove water droplets from the air stream~~. A second bank of HEPA filters follows the adsorber section to collect carbon fines, and provides back-up in case of failure of the main HEPA filter bank.

Also included within a filter train are one air conditioning unit, two kitchen area exhaust duct isolation valves, two toilet area exhaust duct isolation valves and two air intake duct isolation valves in each (north and south) intake duct.

CREVS

f

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, ~~and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident~~. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of Design Basis Accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

air conditioning equipment room, emergency equipment cleanup system equipment room, emergency food and water storage areas, toilet, kitchen/dining/conference room and supervisor's office.

(ITS 5.5.15)

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CREVS

s

The ~~CREACS~~ is an emergency system, part of which ~~may~~ also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the CRE is isolated, and the stream of ventilation air is recirculated through the filter trains of the system. The prefilters ~~and demisters~~ remove any large particles in the air~~, and any entrained water droplets present~~ to prevent excessive loading of the HEPA filters and charcoal adsorbers. ~~Both the demister and heater are important to the effectiveness of the charcoal adsorbers.~~

BASES

BACKGROUND (continued)

CREVS

Actuation of the ~~CREACS~~ places the system into ~~either of two separate states of the emergency mode of operation, depending on the initiation signal. Actuation of the system to the emergency radiation state of the emergency~~ mode of operation closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the redundant trains of HEPA and charcoal ~~filters~~. The ~~emergency radiation state also initiates~~ pressurization and filtered ventilation of the air supply to the CRE.

the recirculation

which

starts control room emergency cleanup fans, standby control room air conditioning unit,

adsorbers

recirculation mode also facilitates

1

Outside air is filtered~~, [diluted with building air from the electrical equipment and cable spreading rooms,]~~ and then added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. ~~The actions taken in the toxic gas isolation state are the same, except that the signal switches the CREACS to an isolation mode, minimizing outside air entering the CRE through the CRE boundary.~~

3

The air entering the CRE is continuously monitored by radiation ~~and toxic gas detectors~~. One detector output above the setpoint will cause actuation of the ~~emergency radiation state or toxic gas isolation state~~ as required. ~~The actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.~~

monitors

recirculation mode

In recirculation mode, a makeup

450

is acceptable to

3

~~A single CREACS train operating at a~~ flow rate of ≤ ~~[~~3000~~]~~ cfm ~~will~~ pressurize the CRE to ~~about [~~0.125~~]~~ inches water gauge relative to ~~external~~ areas adjacent to the CRE boundary, and provides an air exchange rate in excess of 25% per hour. The ~~CREACS~~ operation in maintaining the CRE habitable is discussed in the FSAR, Sections ~~[~~9.4~~]~~ (Ref. 1).

surrounding

CREVS

U

and 6.4 (Ref. 3)

1

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Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation ~~dampers~~ are arranged in series pairs so that the failure of one ~~damper~~ to shut will not result in a breach of isolation. The ~~CREACS~~ is designed in accordance with Seismic Category I requirements.

valves

valve

CREVS

CREVS

The ~~CREACS~~ is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a DBA without exceeding a ~~[5 rem whole body dose or its equivalent to any part of the body]~~ [5 rem total effective dose equivalent (TEDE)~~]~~.

3

BASES

APPLICABLE SAFETY ANALYSES

CREVS The ~~CREACS~~ components are arranged in redundant safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. 1

CREVS The ~~CREACS~~ provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident fission product release presented in the U FSAR, Chapter ~~[~~15~~]~~ (Ref. 2). 1 3

CREVS The ~~CREACS~~ provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 3). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor ~~either~~ from the control room ~~or from the remote shutdown panels (Ref. 4)~~.

CREVS The worst case single active failure of a component of the ~~CREACS~~, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. 1

CREVS The ~~CREACS~~ satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

CREVS Two ~~independent and~~ redundant trains of the ~~CREACS~~ are required to be OPERABLE to ensure that at least one is available, if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of ~~[5 rem whole body or its equivalent to any part of the body]~~ [5 rem TEDE~~]~~ ~~or~~ the CRE occupants in the event of a large radioactive release. 1 3

to

CREVS Each ~~CREACS~~ train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE.

CREVS A ~~CREACS~~ train is considered OPERABLE when the associated:

a. Fan is OPERABLE,

b. HEPA filters and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions, and

c. ~~Heater, demister,~~ ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained. 1

BASES

LCO (continued)

CREVS

In order for the ~~CREACS~~ trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke. 1

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

CREVS

In MODES 1, 2, 3, 4, ~~[~~5, and 6~~]~~, and during movement of ~~[recently]~~ irradiated fuel assemblies, the ~~CREACS~~ must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA. 3

~~In MODES [5 and 6], the CREACS is required to cope with the release from a rupture of an outside waste gas tank.~~ 1

CREVS

During movement of ~~[recently]~~ irradiated fuel assemblies, the ~~CREACS~~ must be OPERABLE to cope with the release from a fuel handling accident ~~[involving handling recently irradiated fuel]~~. ~~[Due to radioactive decay, the CREACS is only required to cope with fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days).]~~ 3 3

ACTIONS

A.1

CREVS

With one ~~CREACS~~ train inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE ~~CREACS~~ train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE ~~CREACS~~ train could result in loss of ~~CREACS~~ function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability. 1

CREVS

CREVS

BASES

ACTIONS (continued)

B.1, B.2, and B.3

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to ~~[5 rem whole body or its equivalent to any part of the body]~~ [5 rem TEDE~~]~~), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days. 3

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

C.1, C.2, and C.3

CREVS

If both ~~CREACS~~ trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable control room boundary (i.e., Condition B), at least one ~~CREACS~~ train must be returned to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second ~~CREACS~~ train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be

CREVS

CREVS

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BASES

ACTIONS (continued)

CREVS
inoperable, or if both trains are discovered to be inoperable at the same time. During the period that the ~~CREACS~~ trains are inoperable, action must be initiated to implement mitigating actions to lessen the effect on
CREVS
CRE occupants from potential hazards while both trains of ~~CREACS~~ are inoperable. In the event of a DBA, the mitigating actions will reduce the consequences of radiological exposures to the CRE occupants.

Specification 3.4.16, "RCS Specific Activity," allows limited operation with the reactor coolant system (RCS) activity significantly greater than the LCO limit. This presents a risk to the plant operator during an accident
CREVS
when all ~~CREACS~~ trains are inoperable. Therefore, it must be verified within 1 hour that LCO 3.4.16 is met. This Required Action does not require additional RCS sampling beyond that normally required by LCO 3.4.16.

CREVS
At least one ~~CREACS~~ train must be returned to OPERABLE status within 24 hours. The Completion Time is based on Reference 3 which demonstrated that the 24 hour Completion Time is acceptable based on the infrequent use of the Required Actions and the small incremental effect on plant risk.

1

D.1 and D.2

~~[~~In MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies, if Required Action A.1 cannot be completed within the
CREVS
required Completion Time, the OPERABLE ~~CREACS~~ train must be
recirculation
immediately placed in the ~~emergency~~ mode of operation. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

3

1

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position.

~~Required Action D.1 is modified by a Note indicating to place the system in the toxic gas protection mode if the automatic transfer to the toxic gas protection mode is inoperable.~~

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BASES

ACTIONS (continued)

E.1

When ~~[~~in MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies, with two ~~CREACS~~ trains inoperable or with one or more ~~CREACS~~ trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

CREVS
3
1

F.1 and F.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

4

CREVS

If the inoperable ~~CREACS~~ or control room boundary cannot be restored to OPERABLE status within the associated Completion Time in MODE 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~5~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

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Required Action F.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable,

BASES

ACTIONS (continued)

because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~11~~.1 (10) (2)

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Operation ~~[with the heaters on]~~ for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that ~~[heater failure,]~~ blockage, fan or motor failure, or excessive vibration can be detected for corrective action. ~~[The 31 day Frequency is based on the known reliability of the equipment, and the two train redundancy available.~~ (3)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ (4) (3)

SURVEILLANCE REQUIREMENTS (continued)

10
2

SR 3.7.~~11~~.2

CREVS
1

This SR verifies that the required ~~CREACS~~ testing is performed in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. The ~~[~~VFTP~~]~~ includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~.
3

(ITS 5.5.8)

10
2

SR 3.7.~~11~~.3

CREVS
1

This SR verifies that each ~~CREACS~~ train starts and operates on an actual or simulated actuation signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. ~~[The Frequency of [18] months is based on industry operating experience and is consistent with the typical refueling cycle.~~
3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------~~~~]~~
4
3

BASES

SURVEILLANCE REQUIREMENTS (continued)

(10) SR 3.7.~~11~~.4 (2)

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than ~~[5 rem whole body or its equivalent to any part of the body]~~ [5 rem TEDE~~]~~ and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. ~~6~~) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. ~~7~~). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. ~~8~~). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

(3) (5) (6) (4) (7)

REFERENCES

1. FSAR, Section ~~[~~9.4~~]~~. (U)
2. FSAR, Chapter ~~[~~15~~]~~. (U)
3. FSAR, Section ~~[~~6.4~~]~~. (U)

(1) (3)

~~4. FSAR, Section [9.5].~~

(4) ~~5~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

(5) ~~6~~. Regulatory Guide 1.196.

(1)

BASES

REFERENCES (continued)

6 ~~7~~. NEI 99-03, "Control Room Habitability Assessment," June 2001.

7 ~~8~~. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040300694).

8 ~~9~~. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

(1)

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.10 BASES, CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS 3.7.11 title "Control Room Emergency Air Cleanup System (CREACS)" has been changed to "Control Room Emergency Ventilation System (CREVS)" consistent with the St. Lucie Plant (PSL) site specific terminology. In addition, PSL design does not include the Essential Chilled Water System (ISTS 3.7.10). Therefore, ISTS 3.7.10 is not included in the PSL ITS. The Control Room Emergency Air Cleanup System (ISTS 3.7.11) is renumbered as ITS 3.7.10.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.10, CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 11

ITS 3.7.11, Control Room Air Conditioning System (CRACS)

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

PLANT SYSTEMS

AIR CONDITIONING

(CRACS)

3/4.7.7 CONTROL ROOM ~~EMERGENCY VENTILATION~~ SYSTEM

LIMITING CONDITION FOR OPERATION

Two CRACS trains

LCO 3.7.11 3.7.7.1 ~~The control room emergency ventilation system~~ shall be OPERABLE ~~with:~~

a. Two booster fans,

b. Two isolation valves in each outside air intake duct,

c. Two isolation valves in the toilet area air exhaust duct,

d. One filter train,

e. ~~At least two air conditioning units, and~~

f. Two isolation valves in the kitchen area exhaust duct.

See ITS 3.7.10

NOTE

The control room envelope boundary may be opened intermittently under administrative control.

Applicability **APPLICABILITY:** MODES 1, 2, 3, 4, 5 and 6 or during movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3 and 4:

a. With one booster fan inoperable, restore the inoperable fan to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

b. With one isolation valve per air duct inoperable, operation may continue provided the other isolation valve in the same duct is maintained closed; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

c. With the filter train inoperable for reasons other than an inoperable Control Room Envelope boundary:

1. Immediately initiate action to implement mitigating actions, and
2. Within 1 hour, verify LCO 3.4.8, "Specific Activity," is met, and
3. Within 24 hours restore the filter train to OPERABLE status.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

See ITS 3.7.10

ACTION A

30

d. ~~With only one air conditioning unit OPERABLE, restore at least two air conditioning units~~ to OPERABLE status within ~~7~~ days or be in at least HOT STANDBY within ~~the next~~ 6 hours and in HOT SHUTDOWN within ~~the following 6~~ hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

L01

ACTION C

12

Required Action C.2 NOTE

One required CRACS train inoperable. Restore CRACS train

PLANT SYSTEMS

ACTION: (continued)

See ITS 3.7.10

MODES 1, 2, 3 and 4: (continued)

NOTE
Action not applicable when second booster fan intentionally made inoperable.

e. With two booster fans inoperable for reasons other than an inoperable Control Room Envelope boundary:

1. Immediately initiate action to implement mitigating actions, and

2. Within 1 hour, verify LCO 3.4.8, “Specific Activity,” is met, and

3. Within 24 hours restore at least one booster fan to OPERABLE status.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Condition B NOTE

second required CRACS train

NOTE
Action not applicable when ~~third air conditioning unit~~ intentionally made inoperable.

two required CRACS trains

ACTION B

ACTION C

f. With ~~three air conditioning units~~ inoperable ~~for reasons other than an inoperable Control Room Envelope boundary,~~ restore at least one ~~air conditioning unit~~ to OPERABLE status within 24 hours or be in ~~at least HOT STANDBY~~ within ~~the next~~ 6 hours and ~~COLD SHUTDOWN~~ within ~~the following 30~~ hours.

12

MODE 3

CRACS train

A02

L02

MODE 4

g. With the filter train inoperable due to an inoperable Control Room Envelope boundary:

1. Immediately initiate actions to implement mitigating actions, and

2. Within 24 hours, verify mitigating actions to ensure Control Room Envelope occupant exposures to radiological, chemical, and smoke hazards will not exceed limits, and

3. Restore Control Room Envelope boundary to OPERABLE status within 90 days.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.7.10

PLANT SYSTEMS

ACTION: (continued)

See ITS 3.7.10

MODES 5 and 6 or during movement of irradiated fuel assemblies:

a. With one booster fan inoperable, restore the inoperable fan to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room emergency ventilation system in the recirculation mode or suspend movement of irradiated fuel assemblies.

b. With one isolation valve in an air duct inoperable, maintain the other isolation valve in the same air duct closed or suspend movement of irradiated fuel assemblies.

c. With the filter train inoperable, suspend movement of irradiated fuel assemblies.

ACTION A

One required CRACS train inoperable. Restore CRACS train

d. ~~With only one air conditioning unit OPERABLE, restore at least two air conditioning units~~ to OPERABLE status within ~~7~~ 30 days or suspend movement of irradiated fuel assemblies.

Required Action D.2

Add proposed Required Action D.1

Add proposed Action E

L01

L03

M01

SURVEILLANCE REQUIREMENTS

CRACS

4.7.7.1 ~~The control room emergency ventilation system~~ shall be demonstrated OPERABLE:

the CRACS has the capability to remove the assumed heat load

SR 3.7.11.1

a. In accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing that the control room air temperature is ≤ 120°F~~.

M02

b. In accordance with the Surveillance Frequency Control Program by:

1. Initiating flow through the HEPA filter and charcoal adsorber train and verifying that each booster fan operates for at least 15 minutes.

2. Starting (unless already operating) each air conditioning unit and verifying that it operates for at least 8 hours.

c. By performing required control room emergency ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

See ITS 3.7.10

PLANT SYSTEMS

AIR CONDITIONING

(CRACS)

3/4.7.7 CONTROL ROOM ~~EMERGENCY AIR CLEANUP~~ SYSTEM ~~(CREACS)~~

LIMITING CONDITION FOR OPERATION

LCO 3.7.11

3.7.7 Two ~~independent control room emergency air cleanup systems~~ shall be OPERABLE ~~with~~:

CRACS trains

a. A filter train and its associated fan per system, and

b. ~~At least one air conditioning unit per system, and~~

c. Two isolation valves in the kitchen area exhaust duct, and

d. Two isolation valves in the toilet area exhaust duct, and

e. Two isolation valves in each (North and South) air intake duct.

See ITS 3.7.10

NOTE

The control room envelope boundary may be opened intermittently under administrative control.

Applicability

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6 or during movement of irradiated fuel assemblies.

ACTION:

Add proposed ACTIONS

M03

MODES 1, 2, 3, and 4:

a. With one control room emergency air cleanup system inoperable for reasons other than an inoperable Control Room Envelope boundary, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.7.10

b. With one or more control room emergency air cleanup systems inoperable due to an inoperable Control Room Envelope boundary:

1. Immediately initiate actions to implement mitigating actions, and
2. Within 24 hours, verify mitigating actions to ensure Control Room Envelope occupant exposures to radiological, chemical, and smoke hazards will not exceed limits, and
3. Restore Control Room Envelope boundary to OPERABLE status within 90 days.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

c. With an isolation valve in an air intake duct or air exhaust duct inoperable, operation may continue provided the other isolation valve in the same air intake or air exhaust duct is maintained closed; otherwise be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

NOTE

Action not applicable when second CREACS train intentionally made inoperable.

d. With two control room emergency air cleanup systems inoperable for reasons other than an inoperable Control Room Envelope boundary:

1. Immediately initiate actions to implement mitigating actions, and
2. Within 1 hour, verify LCO 3.4.8, "Specific Activity," is met, and
3. Within 24 hours restore at least one CREACS train to OPERABLE status.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.7 ~~Each control room emergency air cleanup system~~ CRACS shall be demonstrated OPERABLE:

the CRACS has the capability to remove the assumed heat load

SR 3.7.11.1

a. In accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing that the control room air temperature is ≤ 120°F~~.

M02

b. In accordance with the Surveillance Frequency Control Program by (1) initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes and ~~(2) starting, unless already operating each air conditioning unit and verifying that it operates for at least 8 hours~~.

M02

c. By performing required control room emergency air cleanup system filter testing in accordance with the Ventilation Filter Testing Program.

See ITS 3.7.10

DISCUSSION OF CHANGES
ITS 3.7.11, CONTROL ROOM AIR CONDITIONING SYSTEM (CRACS)

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 1 only:** CTS 3.7.7.1 Action f (MODES 1, 2, 3 and 4) states that with air conditioning units inoperable "for reasons other than an inoperable Control Room Envelope boundary." This excludes any AC unit inoperability due to an inoperable control room envelope boundary. ITS 3.7.11 ACTIONS do not include an exclusion associated with the control room envelope boundary when two required (i.e., three) CRACS trains are inoperable. This changes the CTS by removing the exception associated with the control room envelope boundary when two required CRACS trains are inoperable.

The purpose of the CTS ACTIONS is to clarify that when the control room envelop boundary is breached, the condition related to inoperable AC units does not apply. However, the control room envelop boundary is applicable to the control room filtration function and not the control room cooling function. This change is acceptable because ITS retains the requirements associated with the control room envelope boundary within the Control Room Emergency Ventilation System (CREVS) Technical Specification (ITS 3.7.10) and the Control Room Envelope (CRE) Habitability Program in ITS Section 5.5. This change is designated as administrative and is acceptable because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS 3.7.7.1 does not provide a specific ACTION for the Condition of two required AC units inoperable when in MODES 5 and 6 or during movement of irradiated fuel assemblies. ITS 3.7.11 ACTION E requires the suspension of irradiated fuel assemblies immediately. This changes CTS by adding actions for two inoperable required CRACS trains in MODES 5 and 6 and during movement of irradiated fuel assemblies.

The purpose of ITS 3.7.7.1 ACTION E is to place the unit in a condition that minimizes accident risk. Suspending movement of irradiated fuel assemblies accomplishes this purpose by precluding an event (i.e., fuel handling accident) that could result in a release of radioactivity requiring isolation of the control room. This change is designated as more restrictive because it adds an additional ACTION that CTS did not have.

M02 Unit 1 CTS 4.7.7.1 and Unit 2 CTS 4.7.7.a require verification that the control room air temperature is ≤ 120°F in accordance with the Surveillance Frequency

Control Program. This check is performed as a parameter check during operator rounds. Unit 2 CTS 4.7.7.b requires, in part, starting each air conditioning unit, unless already operating, and verifying that it operates for at least 8 hours. CTS does not provide a specific surveillance that verifies the design capability of the control room AC units. ITS SR 3.7.11.1 requires a verification that the CRACS has the capability to remove the assumed heat load with a Frequency in accordance with the Surveillance Frequency Control Program. This changes the CTS by removing the parameter check of control room air temperature performed as part of operational and procedural processes and adding a specific surveillance to require verifying the capability of the CRACS to remove the design heat load. This also changes Unit 2 CTS by removing an operational consideration to balance AC unit runtime.

The purpose of the CRACS Surveillance is to ensure the equipment in the control room does not exceed the required equipment qualification requirements. Verifying a temperature of < 120°F does not consider heat load profile during the duration of a design basis accident. Operating each Unit 2 AC unit for 8 hours also does not necessarily confirm the capability of the AC unit under accident conditions. The proposed Surveillance verifies the heat removal capability of the CRACS is sufficient to meet design requirements and consists of a combination of testing and calculations. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, SR 3.7.11.1 will be performed at a periodic Frequency in accordance with the SFCP with an initial Frequency of 18 months consistent with ISTS SR 3.7.12.1. This Frequency is appropriate since significant degradation of the CRACS is slow and is not expected over this time period. This change is designated as more restrictive because it adds a Surveillance Requirement that is more comprehensive than the simplistic temperature verification provide in CTS.

M03 **Unit 2 only:** CTS 3.7.7 does not contain specific requirements for the Control Room Air Conditioning System (CRACS). Currently CRACS requirements are encompassed within the Control Room Emergency Air Cleanup System (CREACS)) Operability, Applicability and Actions requirements. ITS 3.7.11 provides a specific Technical Specification that addresses the Control Room Air Conditioning System (CRACS) separately with specific ACTIONS that address the inoperability of the air conditioning units. This changes CTS by incorporating new Technical Specification requirements for the CRACS based on ISTS 3.7.12 (ITS 3.7.11).

The purpose of the CRACS is to maintain the temperature of the control room during normal operations and accident conditions to support control room equipment and maintain temperatures within the respective equipment qualification temperature limits. This change is acceptable because the safety analyses assume the OPERABILITY of the control room equipment. This change is designated as more restrictive because it adds new requirements to CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 **Unit 1 only:** *(Category 3 – Relaxation of Completion Time)* CTS 3.7.7.1 Action d (MODES 1, 2, 3 and 4) and Action d (MODES 5 and 6 or during movement of irradiated fuel assemblies) states "with only one air conditioning unit OPERABLE, restore at least two air conditioning units to OPERABLE status within 7 days." ITS 3.7.11 ACTION A allows 30 days to restore a required inoperable CRACS train to OPERABLE status. This changes the CTS by increasing the time allowed to restore the inoperable components from 7 days to 30 days.

The purpose of CTS 3.7.7.1 is to provide a degree of assurance that the CRACS can provide cooling when required. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the allowed Completion Time. The CRACS train is still required to be restored to OPERABLE status and can perform its function without one air conditioning train. This change is designated as less restrictive because additional time is allowed in the ITS to restore inoperable equipment to within the LCO limits than was allowed in the CTS.

L02 **Unit 1 only:** *(Category 4 – Relaxation of Required Action)* CTS 3.7.7.1 Action f (MODES 1, 2, 3 and 4) requires, when three air conditioning units are inoperable for reasons other than an inoperable control room envelope boundary, to restore at least one air conditioning unit within 24 hours or be in at least HOT STANDBY (MODE 3) with 6 hours and COLD SHUTDOWN (MODE 5) within the following 30 hours. ITS ACTION C requires, for a similar condition, to be in MODE 3 within 6 hours and in MODE 4 within 12 hours with a Note stating that LCO 3.0.4.a is not applicable when entering MODE 4. This changes the CTS by changing the end state of CTS 3.7.7.1 Action f from "COLD SHUTDOWN within the following 30 hours" to "MODE 4 within 12 hours."

The purpose in the CTS action to shutdown the unit when the LCO cannot be restored within the required Completion Time is to place the unit in a MODE that minimizes overall plant risk. Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 as indicated in CE-NPSD-1186-A, Revision 00, "Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs," dated October 2001 (ADAMS Accession No. 110410539). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. Voluntary entry into MODE 5 may continue to be made as it is also an acceptable low-risk state. The change is consistent with similar changes made to CTS 3.7.7.1 in Amendment 234, "St. Lucie Plant, Unit Nos. 1 and 2 – Issuance of

Amendments Regarding Changes in Selected Technical Specifications End States (CAC Nos. MF8106 and MF8107; Formerly CAC NOS. MF6683 AND MF6684)," dated August 30, 2016, (ADAMS Accession No. ML16210A374). This change is acceptable because the modification of the end state from MODE 5 to MODE 4 is consistent with CE-NPSD-1186-A and corrects an inconsistency in the CTS end states for Actions intended to place the unit in a condition outside of the LCO Applicability. Therefore, this change corrects that oversight. This change is designated as less restrictive because instead of requiring the plant to achieve MODE 5 (COLD SHUTDOWN), the ITS end state is MODE 4 (HOT SHUTDOWN) with LCO 3.0.4.a allowance not applicable.

L03 **Unit 1 only:** *(Category 4 – Relaxation of Required Action)* CTS 3.7.7.1 Action d (MODES 5 and 6 or during movement of irradiated fuel assemblies) requires immediate suspension of irradiated fuel assembly movement when a required inoperable air conditioning (AC) unit cannot be restored within the required Completion Time. ITS 3.7.11 includes an optional action, Required Action D.1, to place an OPERABLE AC unit in operation. This changes the CTS by adding an optional action in MODES 5 and 6 and during movement of irradiated fuel assemblies to immediately suspend irradiated fuel assemblies.

The purpose of CTS 3.7.7.1 action in MODES 5, 6, or during movement of irradiated fuel assemblies is to place the plant in a condition where the consequences of a fuel handling accident are precluded or mitigated when the LCO cannot be restored within the allowed Completion Time. The proposed action to place the remaining OPERABLE CRACS train in service ensures the train is OPERABLE and any active failure will be readily detected. This change is acceptable because the optional action is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant train and single active failure to start is precluded by placing the OPERABLE train in service. In addition, the option to immediately suspend irradiated fuel assemblies is retained in the ITS. Therefore, the remedial actions continue to ensure the consequences of a fuel handling accident are precluded or mitigated when the LCO cannot be restored within the required Completion Time. This change is designated as less restrictive because it adds an optional Required Action and Completion Time that CTS does not include.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.7 PLANT SYSTEMS

3.7.~~12~~ Control Room ~~Emergency Air Temperature Control~~ System (~~CREATCS~~)

Air Conditioning
CRACS
11
CRACS
1
2

3.7.7.1 LCO 3.7.~~12~~ Two ~~CREATCS~~ trains shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, 4, ~~[~~5, and 6,~~]~~
During movement of ~~[recently]~~ irradiated fuel assemblies.

3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action d DOC L01 A. One ~~CREATCS~~ train inoperable. (required CRACS)	A.1 Restore ~~CREATCS~~ train to OPERABLE status. (CRACS)	30 days
B. (CRACS) ---------- NOTE ---------- Not applicable when second ~~CREATCS~~ train intentionally made inoperable. ---------- Action f NOTE Two ~~CREATCS~~ trains inoperable in MODE 1, 2, 3, or 4. (required CRACS) Action f DOC A02 DOC L02	B.1 Restore at least one ~~CREATCS~~ train to OPERABLE status. (CRACS)	24 hours
Action d Action f DOC L02 C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

1
2
1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L03 D. Required Action and associated Completion Time of Condition A not met ~~[~~in MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies.	D.1 Place OPERABLE ~~CREATCS~~ train in operation. (CRACS)	Immediately (2)
	OR	
Action d	D.2 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately (3)
DOC M01 E. Two ~~CREATCS~~ trains inoperable ~~[~~in MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies. (required CRACS)	E.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately (2) (3)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.7.7.1.a DOC M02 SR 3.7.~~12~~.1 (11)	Verify ~~each CREATCS train~~ has the capability to remove the assumed heat load. (the CRACS)	~~[[18] months~~ (2) ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

CTS

3.7 PLANT SYSTEMS

3.7.~~12~~ Control Room ~~Emergency Air Temperature Control~~ System (~~CREATCS~~)

Air Conditioning
CRACS
11
CRACS
1 2

3.7.7 LCO 3.7.~~12~~ Two ~~CREATCS~~ trains shall be OPERABLE.

Applicability APPLICABILITY: MODES 1, 2, 3, 4, ~~[~~5, and 6,~~]~~
During movement of ~~[recently]~~ irradiated fuel assemblies.
3

DOC M03 ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~CREATCS~~ train inoperable. (required CRACS)	A.1 Restore ~~CREATCS~~ train to OPERABLE status. (CRACS)	30 days
B. (CRACS) ---------- NOTE ---------- Not applicable when second ~~CREATCS~~ train intentionally made inoperable. ---------- Two ~~CREATCS~~ trains inoperable in MODE 1, 2, 3, or 4. (required CRACS)	B.1 Restore at least one ~~CREATCS~~ train to OPERABLE status. (CRACS)	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

1
2
1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M03 D. Required Action and associated Completion Time of Condition A not met ~~[~~in MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies.	D.1 Place OPERABLE ~~CREATCS~~ train in operation. CRACS OR D.2 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately Immediately 2 3
DOC M03 E. Two ~~CREATCS~~ trains inoperable ~~[~~in MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies. required CRACS	E.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately 2 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.7.7.a DOC M02 SR 3.7.~~12~~.1 11 the CRACS Verify ~~each CREATCS train~~ has the capability to remove the assumed heat load.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ 2 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.11, CONTROL ROOM AIR CONDITIONING SYSTEM (CRACS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS 3.7.12 title "Control Room Emergency Air Temperature Control System (CREATCS)" has been changed to "Control Room Air Conditioning System (CRACS)" consistent with the St. Lucie Plant (PSL) site specific terminology. In addition, PSL design does not include the Essential Chilled Water System (ISTS 3.7.10). Therefore, ISTS 3.7.10 is not included in the PSL ITS. The Control Room Emergency Air Temperature Control System (ISTS 3.6.12) is renumbered as ITS 3.7.11.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.~~12~~ Control Room ~~Emergency Air Temperature Control~~ System (~~CREATCS~~)

Air Conditioning
CRACS
11
1
2

BASES

Insert 1

BACKGROUND

The ~~CREATCS~~ provides temperature control for the control room ~~following isolation of the control room.~~

CRACS
three
Insert 2

The ~~CREATCS~~ consists of ~~two independent,~~ redundant trains that provide cooling ~~and heating~~ of recirculated control room air. Each train consists of ~~heating coils,~~ cooling coils, instrumentation, and controls to provide for control room temperature control.

CRACS
s

The ~~CREATCS~~ is a~~n emergency~~ system~~, parts of~~ which ~~may~~ also operate during normal unit operations. A single train will provide the required temperature control to maintain the control room ~~between [70]°F and [85]~~°F. The ~~CREATCS~~ operation to maintain the control room temperature is discussed in the FSAR, Section ~~[~~6.4~~]~~ (Ref. 1).

≤ 75
CRACS
s
U
and 9.4 (Ref. 2)
1
3
3

APPLICABLE SAFETY ANALYSES

CRACS
ambient

The design basis of the ~~CREATCS~~ is to maintain temperature of the control room ~~environment throughout 30 days of continuous occupancy.~~

Insert 3

CRACS

The ~~CREATCS~~ components are arranged in redundant ~~safety related~~ trains. During emergency operation, the ~~CREATCS~~ maintains the temperature ~~between [70]°F and [85]~~°F. A single active failure of a component of the ~~CREATCS~~, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The ~~CREATCS~~ is designed in accordance with Seismic Category I requirements. The ~~CREATCS~~ is capable of removing sensible and latent heat loads from the control room, considering equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

≤ 125

The ~~CREATCS~~ satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

3
1

LCO

Two independent and redundant trains of the ~~CREATCS~~ are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

Insert 4

CRACS

The ~~CREATCS~~ is considered OPERABLE when the individual components that are necessary to maintain the control room temperature are OPERABLE in both trains. These components include the cooling coils and associated temperature control instrumentation. In addition, the ~~CREATCS~~ must be OPERABLE to the extent that air circulation can be maintained.

1

Insert 1 (1)

is a subsystem of the Control Room Ventilation System and

Insert 2 (1)

during normal operations and accident conditions

Insert 3 (1)

for equipment qualification. Since equipment within the control room is qualified for higher temperature, loss of all air conditioning units is acceptable. At 125°F, continued habitability for periods of 2 hours is permissible. Through judicious allocation of plant operating personnel it is possible to maintain continuous occupancy of the control room.

Insert 4

(1)

Electrical independence requires that the two trains be powered from separate emergency buses.

BASES

APPLICABILITY

CRACS

In MODES 1, 2, 3, 4, ~~[~~5, and 6,~~]~~ and during movement of ~~[recently]~~ irradiated fuel assemblies ~~[(i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~, the ~~CREATCS~~ must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY requirements following isolation of the control room. 3 1

~~In MODES 5 and 6, CREATCS may not be required for those facilities which do not require automatic control room isolation.~~ 1

ACTIONS

A.1

required

With one ~~CREATCS~~ train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE ~~CREATCS~~ train is adequate to maintain the control room temperature within limits. The 30 day Completion Time is reasonable, based on the low probability of an event occurring requiring control room isolation, consideration that the remaining train can provide the required capabilities, and the alternate safety or nonsafety related cooling means that are available. 1

CRACS

B.1

CRACS

required

If two ~~CREATCS~~ trains are inoperable, at least one ~~CREATCS~~ train must be returned to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second ~~CREATCS~~ train is intentionally ~~declared~~ inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. The Completion Time is based on Reference ~~2~~ 3 which demonstrated that the 24 hour Completion Time is acceptable based on the infrequent use of the Required Action and the small incremental effect on plant risk. 1

CRACS

made

C.1 and C.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~ 4

BASES

ACTIONS (continued)

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~ 4

required CRACS
In MODE 1, 2, 3, or 4, when one or more ~~CREATCS~~ trains cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes overall plant risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. 1

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~2~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 3 1

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

~~[~~ D.1 and D.2

CRACS
In MODE 5 or 6, or during movement of ~~[recently]~~ irradiated fuel assemblies, when Required Action A.1 cannot be completed within the required Completion Time, ~~the~~ OPERABLE ~~CREATCS~~ train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE~~, that no failures preventing automatic actuation will occur,~~ and that any active failure will be readily detected. 3

an

BASES

ACTIONS (continued)

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position.~~]~~ 3

~~[~~ E.1

In ~~[~~MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies, with two ~~CREATCS~~ trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.~~]~~ 3

required
CRACS

SURVEILLANCE REQUIREMENTS

SR 3.7.~~12~~.1 11

This SR verifies that the heat removal capability of the system is sufficient to meet design requirements. This SR consists of a combination of testing and calculations. ~~[An [18] month Frequency is appropriate, since significant degradation of the CREATCS is slow and is not expected over this time period.~~ 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4 3

REFERENCES

U

1. FSAR, Section ~~[~~6.4~~]~~. 3
2. UFSAR, Section 9.4
3. ~~2~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001. 1
4. ~~3~~. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

B 3.7 PLANT SYSTEMS

B 3.7.12 Control Room Emergency Air Temperature Control System (CREATCS)

BASES

BACKGROUND

The CREATCS provides temperature control for the control room following isolation of the control room.

The CREATCS consists of two independent, redundant trains that provide cooling and heating of recirculated control room air. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for control room temperature control.

The CREATCS is an emergency system, parts of which may also operate during normal unit operations. A single train will provide the required temperature control to maintain the control room between [70]°F and [85]°F. The CREATCS operation to maintain the control room temperature is discussed in the FSAR, Section [6.4] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CREATCS is to maintain temperature of the control room environment throughout 30 days of continuous occupancy.

The CREATCS components are arranged in redundant safety related trains. During emergency operation, the CREATCS maintains the temperature between [70]°F and [85]°F. A single active failure of a component of the CREATCS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The CREATCS is designed in accordance with Seismic Category I requirements. The CREATCS is capable of removing sensible and latent heat loads from the control room, considering equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

The CREATCS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant trains of the CREATCS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CREATCS is considered OPERABLE when the individual components that are necessary to maintain the control room temperature are OPERABLE in both trains. These components include the cooling coils and associated temperature control instrumentation. In addition, the CREATCS must be OPERABLE to the extent that air circulation can be maintained.

Insert 1 (1)

is a subsystem of the Control Room Ventilation System and

Insert 2 (1)

during normal operations and accident conditions

Insert 3 (1)

for equipment qualification. CRACS is designed to control the environment assuming the simultaneous operation of all equipment located in the control room envelope coinciding with the highest site design ambient conditions. Under these assumptions, the system is capable of maintaining the environmental conditions of 76°F to 84°F and less than 70 percent relative humidity during an accident condition.

Insert 4 (1)

Electrical independence requires that the two trains be powered from separate emergency buses.

BASES

APPLICABILITY

CRACS

In MODES 1, 2, 3, 4, ~~[~~5, and 6,~~]~~ and during movement of ~~[recently]~~ irradiated fuel assemblies ~~[(i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~, the ~~CREATCS~~ must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY requirements following isolation of the control room. 3 1

~~In MODES 5 and 6, CREATCS may not be required for those facilities which do not require automatic control room isolation.~~ 1

ACTIONS

A.1

required

With one ~~CREATCS~~ train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE ~~CREATCS~~ train is adequate to maintain the control room temperature within limits. The 30 day Completion Time is reasonable, based on the low probability of an event occurring requiring control room isolation, consideration that the remaining train can provide the required capabilities, and the alternate safety or nonsafety related cooling means that are available. 1

CRACS

B.1

CRACS

required

If two ~~CREATCS~~ trains are inoperable, at least one ~~CREATCS~~ train must be returned to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second ~~CREATCS~~ train is intentionally ~~declared~~ inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. The Completion Time is based on Reference ~~2~~ 3 which demonstrated that the 24 hour Completion Time is acceptable based on the infrequent use of the Required Action and the small incremental effect on plant risk. 1

CRACS

made

3

C.1 and C.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~ 4

BASES

ACTIONS (continued)

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~ 4

required CRACS
In MODE 1, 2, 3, or 4, when one or more ~~CREATCS~~ trains cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes overall plant risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. 1

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~2~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 3 1

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

~~[~~ D.1 and D.2

CRACS
In MODE 5 or 6, or during movement of ~~[recently]~~ irradiated fuel assemblies, when Required Action A.1 cannot be completed within the required Completion Time, ~~the~~ OPERABLE ~~CREATCS~~ train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE~~, that no failures preventing automatic actuation will occur,~~ and that any active failure will be readily detected. 3

an

BASES

ACTIONS (continued)

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position.~~]~~ 3

~~[~~ E.1

In ~~[~~MODE 5 or 6, or~~]~~ during movement of ~~[recently]~~ irradiated fuel assemblies, with two ~~CREATCS~~ trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.~~]~~ 3

required
CRACS

SURVEILLANCE REQUIREMENTS

SR 3.7.~~12~~.1 11

This SR verifies that the heat removal capability of the system is sufficient to meet design requirements. This SR consists of a combination of testing and calculations. ~~[An [18] month Frequency is appropriate, since significant degradation of the CREATCS is slow and is not expected over this time period.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 3

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4 3

REFERENCES

U

1. FSAR, Section ~~[~~6.4~~]~~. 3
2. UFSAR, Section 9.4
3. ~~2~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001. 1
4. ~~3~~. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.11 BASES, CONTROL ROOM AIR CONDITIONING SYSTEM (CRACS)

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS 3.7.12 title "Control Room Emergency Air Temperature Control System (CREATCS)" has been changed to "Control Room Air Conditioning System (CRACS)" consistent with the St. Lucie Plant (PSL) site specific terminology. In addition, PSL design does not include the Essential Chilled Water System (ISTS 3.7.10). Therefore, ISTS 3.7.10 is not included in the PSL ITS. The Control Room Emergency Air Temperature Control System (ISTS 3.6.12) is renumbered as ITS 3.7.11.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant information/value is inserted to reflect the current licensing basis.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.11, CONTROL ROOM AIR CONDITIONING SYSTEM (CRACS)

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 12

3.7.12, Emergency Core Cooling System (ECCS) Area Ventilation System

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

PLANT SYSTEMS

3/4.7.8 ECCS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

LA01

LCO 3.7.12 ~~3.7.8.1~~ Two ~~independent~~ ECCS area ~~exhaust air filter~~ trains shall be OPERABLE.

ventilation

Add proposed LCO Note — L01

Applicability **APPLICABILITY**: MODES 1, 2, 3 and 4.

ACTION:

ventilation | MODE 3 | ECCS area ventilation

ACTION A ~~With~~ one ECCS area ~~exhaust air filter~~ train inoperable, restore ~~the~~ ~~inoperable~~ train to OPERABLE status within 7 days or be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT SHUTDOWN~~ within ~~the following~~ ~~6~~ hours.

ACTION A.1

ACTION D LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION D.2 Note

MODE 4 | 12

Add proposed ACTION B — L02

Add proposed ACTION C — L02

SURVEILLANCE REQUIREMENTS

~~4.7.8.1 Each ECCS area exhaust air filter train shall be demonstrated OPERABLE:~~

~~a.~~ In accordance with the Surveillance Frequency Control Program ~~by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the~~ train ~~operates~~ for ~~at least~~ 15 minutes.

LA02

SR 3.7.12.1

Operate each ECCS area ventilation | ≥ | continuous

SR 3.7.12.2 ~~b.~~ ~~By~~ perform~~ing~~ required ECCS area ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

(VFTP)

~~c.~~ In accordance with the Surveillance Frequency Control Program:

~~1.~~ Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.

See ITS 5.5.8

actuates on an actual or simulated — L03

SR 3.7.12.3 ~~2.~~ Verify~~ing that the filter~~ train ~~starts on a Safety Injection~~ Actuation Signal.

each ECCS area ventilation

except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position — L04

Add proposed SR 3.7.12.4 — M01

A01

~~Page Deleted~~

A01

~~Page Deleted~~

PLANT SYSTEMS

3/4.7.8 ECCS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.7.12 ~~3.7.8~~ Two ~~independent~~ ECCS area ventilation ~~systems~~ shall be OPERABLE.

LA01

trains

Add proposed LCO Note

L01

Applicability **APPLICABILITY**: MODES 1, 2, 3, and 4.

ACTION:

train

MODE 3

ECCS area ventilation train

ACTION A ~~With~~ one ECCS area ventilation ~~system~~ inoperable, restore ~~the inoperable system~~ to OPERABLE

ACTION A.1 status within 7 days or be in ~~at least~~ ~~HOT STANDBY~~ within ~~the next~~ 6 hours and in ~~HOT~~

ACTION D ~~SHUTDOWN~~ within ~~the following~~ ~~6~~ hours. LCO 3.0.4.a is not applicable when entering HOT

ACTION D.2 Note SHUTDOWN.

MODE 4

12

Add proposed ACTION B

L02

Add proposed ACTION C

L02

SURVEILLANCE REQUIREMENTS

~~4.7.8~~ ~~Each ECCS area ventilation system shall be demonstrated OPERABLE:~~

~~a.~~ In accordance with the Surveillance Frequency Control Program ~~by initiating from the control room and verifying that the system operates~~ for ~~at least~~ 15 minutes.

LA02

SR 3.7.12.1

≥

continuous

Operate each ECCS area ventilation train

SR 3.7.12.2 ~~b.~~ ~~By~~ perform~~ing~~ required ECCS area ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

(VFTP)

SR 3.7.12.3 ~~c.~~ In accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing that the system starts on a~~ safety injection actuation ~~test~~ signal.

each ECCS area ventilation train

except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position

L04

actuates on an actual or simulated

L03

Add proposed SR 3.7.12.4

M01

DISCUSSION OF CHANGES
ITS 3.7.12, ECCS AREA VENTILATION SYSTEM

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 ITS SR 3.7.12.4 states "Verify one ECCS area ventilation train can maintain a negative pressure relative to adjacent areas during the post accident mode of operation at a flow rate of < 27,000 cfm." CTS does not require this Surveillance. This changes CTS by adding a requirement that equipment be able to provide a negative pressure relative to adjacent areas inside the ECCS pump room boundary. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, SR 3.7.12.4 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 18 months on a staggered test basis consistent with the ISTS SR 3.7.12.4 Frequency.

The purpose of ITS SR 3.7.12.4 is to provide assurance that the ECCS pump room boundary can support the function of the ECCS Area Ventilation System of maintaining a negative pressure inside the ECCS pump room boundary while filtering air discharged from those areas to prevent unfiltered LEAKAGE. This change is necessary to ensure sufficient testing is conducted at a frequency to assure that the necessary quality of systems and components is maintained, and that the ECCS Area Ventilation System LCO will be met. This SR verifies the integrity of the ECCS pump room boundary.

The performance Frequency of ITS SR 3.7.12.4 is in accordance with the Surveillance Frequency Control Program (SFCP). PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). The initial periodic Frequency established in accordance with the SFCP will be 18 months on a staggered test basis. See FPL (PSL Unit 1 and Unit 2) “Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program” (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066), which approved the adoption of a SFCP at PSL. This change is designated as more restrictive because a Surveillance Requirement is added to the Technical Specifications.

DISCUSSION OF CHANGES
ITS 3.7.12, ECCS AREA VENTILATION SYSTEM

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.7.8.1 and Unit 2 CTS 3.7.8 state that two "independent" ECCS area ventilation systems shall be OPERABLE. ITS LCO 3.7.12 states that two ECCS area ventilation trains shall be OPERABLE. This changes the CTS by removing details that the trains are "independent" from the CTS to the Bases.

The removal of these details, which are related to the system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that two ECCS area ventilation trains shall be OPERABLE. The design detail that the trains are independent do not need to appear in the Specification in order for the requirement to apply. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS 4.7.8.1.a and Unit 2 CTS 4.7.8.a state, in part, that each ECCS area ventilation train shall be demonstrated OPERABLE by “initiating, from the control room, and verifying that the system operates for at least 15 minutes.” ITS SR 3.7.12.1 states "Operate each ECCS area ventilation train for ≥ 15 continuous minutes." This changes the CTS by moving the requirement to actuate the train from the control room to the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to operate the ECCS area ventilation trains for ≥ 15 continuous minutes. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* Unit 1 CTS 3.7.8.1 and Unit 2 CTS 3.7.8 requires two ECCS area ventilation trains to be OPERABLE. ITS LCO 3.7.12 includes the same ECCS area ventilation trains OPERABILITY requirement but is modified by a Note, which states "The ECCS pump room boundary may be opened intermittently under administrative control." This changes the CTS by allowing the ECCS pump room boundary to be opened under administrative control when the ECCS area ventilation trains are required to be OPERABLE.

The purpose of the Note to ITS LCO 3.7.12 is to allow the ECCS pump room negative pressure boundary to be opened under administrative control. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. The ITS allowance requires administrative controls to be in place in order to open the boundary. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for ECCS pump room negative pressure boundary isolation is indicated. These administrative controls minimize risk by providing compensatory actions that ensure the boundary can be rapidly restored to the design condition. The required administrative controls are described in the Bases. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.7.8.1 and Unit 2 CTS 3.7.8 Action contain compensatory actions to take when one ECCS area ventilation train is inoperable. The Unit 1 CTS 3.7.8.1 and Unit 2 CTS 3.7.8 Action does not contain compensatory actions to take when two ECCS area ventilation trains are inoperable. Therefore, CTS 3.0.3 would be entered for two ECCS area ventilation trains inoperable. CTS 3.0.3 requires action to be initiated within one hour to be in HOT STANDBY (ITS MODE 3) in the following 6 hours, to be in HOT SHUTDOWN (ITS MODE 4) in the following 6 hours, and to be in COLD SHUTDOWN (ITS MODE 5) in the subsequent 36 hours. ITS 3.7.12 ACTION B states, with two ECCS area ventilation trains inoperable due to an inoperable ECCS pump room boundary, to restore the ECCS pump room boundary to OPERABLE status within 24 hours. Additionally, ITS 3.7.12 ACTION C requires, when two ECCS area ventilation trains are unintentionally inoperable for reasons other than Condition B (i.e., an inoperable ECCS pump room boundary), one ECCS area ventilation train to be restored to OPERABLE status within 24 hours. This changes the CTS by not requiring entry into LCO 3.0.3 and allowing 24 hours to restore the ECCS pump room boundary to OPERABLE status and allowing 24 hours to restore at least one ECCS area ventilation train to OPERABLE status when two ECCS area ventilation trains are unintentionally inoperable for reasons other than Condition B.

ITS 3.7.12 ACTION B allows 24 hours to restore the pressure ECCS pump room boundary to OPERABLE status before requiring an orderly shutdown from operating conditions. This change is acceptable because this change would allow for routine repairs. Compensatory measures will be taken to protect plant personnel from potential hazards, and preplanned compensatory measures will be in place to address both the intentional and unintentional inoperability of the ECCS pump room boundary. Furthermore, the 24 hour Completion Time is based on the low probability of a DBA occurring during this time period and the compensatory measures that will be taken.

The addition of ITS 3.7.12 ACTION B is consistent with the Revision 5 of the ISTS as adopted by Technical Specifications Task Force (TSTF) traveler TSTF-287-A, Revision 5, "Ventilation System Envelope Allowed Outage Time" dated March 16, 2000. As stated in the Unit 1 and Unit 2 UFSAR, Section 9.4.3.1, the ECCS Area Ventilation System is sized to maintain a slightly negative pressure in the ECCS pump room area with respect to surrounding areas of the reactor auxiliary building. Access into the ECCS pump room area from other areas of the reactor auxiliary building is through gasketed self-closing or locked closed doors. Opening of locked doors is controlled under administrative controls.

ITS 3.7.12 ACTION C allows 24 hours to restore at least one ECCS area ventilation trains to OPERABLE status before requiring an orderly shutdown from operating conditions. This change is acceptable because overall safety is enhanced by avoiding an unnecessary plant shutdown and minimizing plant transients and associated transitions and realignment risks. The Condition is modified by a Note stating it is not applicable if the second ECCS area ventilation train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. ITS 3.7.12, Condition C is only applicable if one ECCS area ventilation train is inoperable for any reason and the second ECCS area ventilation train is discovered to be inoperable, or if both ECCS area ventilation trains are discovered to be inoperable at the same time.

The addition of ITS 3.7.12 ACTION C is consistent with Revision 5 of the ISTS as adopted by TSTF traveler TSTF-426-A, Revision 5, "Revise or Add Actions to Preclude Entry into LCO 3.0.3 - RITSTF Initiatives 6b & 6c" dated May 30, 2013. The TSTF was approved for licensee adoption as documented in Federal Register Notice *78 FR 32476* (NRC ADAMS Accession No. ML13036A381). The PSL Unit 1 and Unit 2 ECCS Area Ventilation System design does not rely upon the system to provide temperature control of the ECCS pump rooms, and therefore, does not affect OPERABILITY of the ECCS pumps. The reactor auxiliary building ventilation systems are designed to provide ventilation to permit proper functioning of equipment during normal operation.

Additionally, ITS 3.7.12 ACTION D continues to require the unit to be placed outside of the MODE of Applicability when two ECCS area ventilation trains are inoperable if any Required Action of Condition A, B, or C cannot be met with the associated Completion Time. LCO 3.0.3 continues to apply when two ECCS area ventilation trains are intentionally made inoperable for reasons other than an inoperable ECCS pump room boundary. This change is designated as less

restrictive because the less stringent requirements are being applied in the ITS than were applied in the CTS.

L03 *(Category 6 - Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.7.8.1.c.2 and Unit 2 CTS 4.7.8.c require verification of the automatic actuation of ECCS area ventilation trains on a safety injection actuation signal (Unit 1) and safety injection actuation “test” signal (Unit 2). ITS SR 3.7.12.3 specifies that the signal may be from either an "actual" or simulated (i.e., test) signal. This changes the CTS by explicitly allowing the use of either an actual or simulated signal for the test.

The purpose of the ECCS area ventilation train actuation test is to ensure the ECCS area ventilation trains operate correctly upon receipt of an actuation signal. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual," "simulated," or "test" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L04 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.7.8.1.c.2 and Unit 2 CTS 4.7.8.c require verification of the automatic actuation of ECCS area ventilation trains on a safety injection actuation signal (Unit 1) and safety injection actuation “test” signal (Unit 2). ITS SR 3.7.12.3 requires verification that each ECCS area ventilation train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position. This changes the CTS by excluding those ECCS area ventilation automatic dampers and valves that are locked, sealed, or otherwise secured in position from the verification.

The purpose of the ECCS area ventilation train actuation test is to provide assurance that if an event occurred requiring the ECCS area ventilation dampers and valves to be in their correct position, those requiring automatic actuation would actuate to their correct position. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Those automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position are not required to actuate on a safety injection actuation signal in order to perform their safety function because they are already in the required position. Testing such valves would not provide any additional assurance of OPERABILITY. Valves and dampers that are required to actuate will continue to be tested. In addition, restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. This change

is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7.~~13~~ Emergency Core Cooling System (ECCS) ~~Pump Room Exhaust Air Cleanup~~ System ~~(PREACS)~~

Area Ventilation
1
3
12

area ventilation
1
3

3.7.8.1 LCO 3.7.~~13~~

Two ECCS ~~PREACS~~ trains shall be OPERABLE.

----------NOTE----------

DOC L01 The ECCS pump room boundary may be opened intermittently under administrative control.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.8.1 Action A. One ECCS ~~PREACS~~ train inoperable.	A.1 Restore ECCS ~~PREACS~~ train to OPERABLE status.	7 days
DOC L02 B. Two ECCS ~~PREACS~~ trains inoperable due to inoperable ECCS pump room boundary.	B.1 Restore ECCS pump room boundary to OPERABLE status.	24 hours
DOC L02 C. ---------- NOTE ---------- Not applicable when second ECCS ~~PREACS~~ train intentionally made inoperable. ---------- Two ECCS ~~PREACS~~ trains inoperable for reasons other than Condition B.	C.1 Restore at least one ECCS ~~PREACS~~ train to OPERABLE status.	24 hours

area ventilation
area ventilation
1
area ventilation
area ventilation
1
area ventilation
1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.8.1 Action D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND	6 hours
3.7.8.1 Action	D.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.7.8.1.a	SR 3.7.~~13~~.1 (12) Operate each ECCS ~~PREACS~~ train (area ventilation) for ≥ 15 continuous minutes ~~[with heater operating]~~.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
4.7.8.1.b	SR 3.7.~~13~~.2 (12) Perform required ECCS ~~PREACS~~ (area ventilation) filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~.	In accordance with the ~~[~~VFTP~~]~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.7.8.1.c.2 DOC L03 DOC L04	SR 3.7.~~13~~.3	Verify each ECCS ~~PREACS~~ train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 1 2
DOC M01	SR 3.7.~~13~~.4	Verify one ECCS ~~PREACS~~ train can maintain a negative pressure ~~≥ [ ] inches water gauge~~ relative to atmospheric pressure during the ~~[~~post accident~~]~~ mode of operation at a flow rate of ≤ ~~[20,000]~~ 27,000 cfm.	~~[[18] months on a STAGGERED TEST BASIS~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 1 2 2 2
	~~SR 3.7.13.5~~	~~[Verify each ECCS PREACS filter bypass damper can be opened, except for dampers that are locked, sealed, or otherwise secured in the open position.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~	4

3.7 PLANT SYSTEMS

3.7.~~13~~ Emergency Core Cooling System (ECCS) ~~Pump Room Exhaust Air Cleanup~~ System ~~(PREACS)~~

Area Ventilation
1
3
12

area ventilation
1
3

3.7.8 LCO 3.7.~~13~~ Two ECCS ~~PREACS~~ trains shall be OPERABLE.

12

----------NOTE----------

DOC L01 The ECCS pump room boundary may be opened intermittently under administrative control.

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
area ventilation 3.7.8 Action A. One ECCS ~~PREACS~~ train inoperable.	area ventilation A.1 Restore ECCS ~~PREACS~~ train to OPERABLE status.	7 days
area ventilation DOC L02 B. Two ECCS ~~PREACS~~ trains inoperable due to inoperable ECCS pump room boundary.	B.1 Restore ECCS pump room boundary to OPERABLE status.	24 hours
area ventilation DOC L02 C. ---------- NOTE ---------- Not applicable when second ECCS ~~PREACS~~ train intentionally made inoperable. ---------- area ventilation Two ECCS ~~PREACS~~ trains inoperable for reasons other than Condition B.	area ventilation C.1 Restore at least one ECCS ~~PREACS~~ train to OPERABLE status.	24 hours

1
1
1
1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.8 Action D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	AND	
3.7.8 Action	D.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.7.8.a SR 3.7.~~13~~.1	Operate each ECCS ~~PREACS~~ train for ≥ 15 continuous minutes ~~[with heater operating]~~.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
4.7.8.b SR 3.7.~~13~~.2	Perform required ECCS ~~PREACS~~ filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~.	In accordance with the ~~[~~VFTP~~]~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.~~13~~.3	Verify each ECCS ~~PREACS~~ train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
SR 3.7.~~13~~.4	Verify one ECCS ~~PREACS~~ train can maintain a negative pressure ~~≥ [ ] inches water gauge~~ relative to atmospheric pressure during the ~~[~~post accident~~]~~ mode of operation at a flow rate of ≤ ~~[20,000]~~ 27,000 cfm.	~~[[18] months on a STAGGERED TEST BASIS~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
~~SR 3.7.13.5~~	~~[Verify each ECCS PREACS filter bypass damper can be opened, except for dampers that are locked, sealed, or otherwise secured in the open position.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

4.7.8.c
DOC L03
DOC L04
DOC M01

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.12, ECCS AREA VENTILATION SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS Specification number and title.

4. As stated in the Unit 1 and Unit 2 UFSAR, Section 9.4.3.1, the ECCS Area Ventilation System is sized to maintain a slightly negative pressure in the ECCS area with respect to surrounding areas of the reactor auxiliary building. Access into the ECCS pump room area from other areas of the reactor auxiliary building is through gasketed self-closing or locked closed doors. Opening of locked doors is controlled under administrative controls.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

Area Ventilation System
B 3.7.~~13~~ Emergency Core Cooling System (ECCS) ~~Pump Room Exhaust Air Cleanup System (PREACS)~~

BASES

Area Ventilation System
BACKGROUND The ECCS ~~PREACS~~ filters air from the area of the active ECCS components during the recirculation phase of a loss of coolant accident (LOCA). The ECCS ~~PREACS~~, in conjunction with other, normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area and the lower reaches of the Auxiliary Building. R1

Area Ventilation System
Reactor

Area Ventilation System
The ECCS ~~PREACS~~ consists of two independent and redundant trains. Each train consists of ~~a heater,~~ a prefilter ~~or demister~~, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system~~, as well as demisters functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines and to back up the upstream HEPA filter, should it develop a leak~~. The system initiates filtered ventilation of the pump room and lower region of the Auxiliary Building following receipt of a safety injection actuation signal ~~or coolant injection actuation signal~~.

Reactor

Area Ventilation System
The ECCS ~~PREACS~~ is a standby system, parts of which may also operate during normal unit operations. The Reactor Auxiliary Building Main Ventilation System provides normal cooling. During emergency operations, the ECCS ~~PREACS~~ dampers are realigned and fans are started to initiate filtration. Upon receipt of ~~the actuating Engineered Safety Feature Actuation System~~ signal~~(s)~~, normal air discharges from the ECCS pump room, the pump room is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters ~~or demisters~~ remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

Area Ventilation System
a safety injection actuation

Area Ventilation System
U
6.3.1
9.4.3
The ECCS ~~PREACS~~ is discussed in the FSAR, Sections ~~[6.5.1]~~, ~~[9.4.5]~~, and ~~[15.6.5]~~ (Refs. 1, 2, and 3, respectively), as it may be used for normal, as well as post accident, atmospheric cleanup functions. ~~The primary purpose of the heaters is to maintain the relative humidity at an acceptable level consistent with iodine removal efficiencies, as discussed in the Regulatory Guide 1.52 (Ref. 4).~~

15.4.1

BASES

Area Ventilation System

APPLICABLE SAFETY ANALYSES

The design basis of the ECCS ~~PREACS~~ is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as safety injection pump seal failure, during the recirculation mode. In such a case, the system limits the radioactive release to within ~~10 CFR 100~~ limits (Ref. ~~5~~); ~~or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits)~~. The analysis of the effects and consequences of a large break LOCA is presented in Reference 3. The ECCS ~~PREACS~~ also actuates following a small break LOCA, requiring the unit to go into the recirculation mode of long term cooling and to clean up releases of smaller leaks, such as from valve stem packing.

10 CFR 50.67

4

Area Ventilation System

The two types of system failures that are considered in the accident analysis are complete loss of function and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

Area Ventilation System

The ECCS ~~PREACS~~ satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

area ventilation

LCO

Two independent and redundant ECCS ~~PREACS~~ trains are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the ECCS pump room exceeding the required limits in the event of a Design Basis Accident (DBA).

Area Ventilation System

ECCS ~~PREACS~~ is considered OPERABLE when the individual components necessary to maintain the ECCS Pump Room filtration are OPERABLE in both trains.

area ventilation

An ECCS ~~PREACS~~ train is considered OPERABLE when its associated:

a. Fan is OPERABLE,

b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions, and

c. ~~Heater, demister,~~ ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

BASES

LCO (continued)

The LCO is modified by a Note allowing the ECCS pump room boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for ECCS pump room isolation is indicated.

APPLICABILITY

Area Ventilation System

In MODES 1, 2, 3, and 4, the ECCS ~~PREACS~~ is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS. 1

Area Ventilation System

In MODES 5 and 6, the ECCS ~~PREACS~~ is not required to be OPERABLE, since the ECCS is not required to be OPERABLE. 1

ACTIONS

A.1

area ventilation

Area Ventilation System

With one ECCS ~~PREACS~~ train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS ~~PREACS~~ function. 1 1

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time) and this system is not a direct support system for the ECCS. The 7 day Completion Time is reasonable, based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.

B.1

~~REVIEWER'S NOTE~~

~~Adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of an intentional and unintentional entry into Condition B.~~ 4

BASES

ACTIONS (continued)

area ventilation

If the ECCS pump room boundary is inoperable, the ECCS ~~PREACS~~ trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ECCS pump room boundary within 24 hours. During the period that the ECCS pump room boundary is inoperable, appropriate compensatory measures ~~[~~consistent with the intent, as applicable, of GDC 19~~, 60, 64 and 10 CFR Part 100]~~ should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ECCS pump room boundary.

area ventilation system

~~REVIEWER'S NOTE~~

~~Condition C is not applicable to plant designs that rely on the ECCS PREACS to provide temperature control of the ECCS pump rooms. Reference 6 did not evaluate the effect of two inoperable ECCS PREACS trains on the ECCS pump room cooling function.~~

C.1

area ventilation

With two ECCS ~~PREACS~~ trains inoperable for reasons other than an inoperable boundary, action must be taken to restore at least one ECCS ~~PREACS~~ train to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second ECCS ~~PREACS~~ train is intentionally declared inoperable. The Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one train is inoperable for any reason and the second train is found to be inoperable, or if both trains are found to be inoperable at the same time. The Completion Time is based on Reference ~~6~~ which demonstrated that the 24 hour Completion Time is acceptable based on the low frequency of the potential challenge and the small incremental risk associated with continued operation.

area ventilation

BASES

ACTIONS (continued)

D.1 and D.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

2

area ventilation

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If the ECCS ~~PREACS~~ train or ECCS pump room boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be in a MODE in which overall plant risk is minimized. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

6

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~7~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

1

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.~~13~~.1

12

periodically with initiation from the control room

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train ~~once a month~~ provides an adequate check on this system. Operation ~~[with the heaters on]~~ for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that ~~[heater failure,]~~ blockage, fan or motor failure, or excessive vibration can be detected for corrective action. ~~[The 31 day Frequency is based on the known reliability of equipment, and the two train redundancy available.~~

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1
2
2
2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

2

SR 3.7.~~13~~.2

12

Area Ventilation System

This SR verifies that the required ECCS ~~PREACS~~ testing is performed in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. The ~~[~~VFTP~~]~~ includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~.

3
1
2

SR 3.7.~~13~~.3

12

area ventilation

This SR verifies that each ECCS ~~PREACS~~ train starts and operates on an actual or simulated actuation signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position

3
1

BASES

SURVEILLANCE REQUIREMENTS (continued)

prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. ~~[The [18] month Frequency is consistent with that specified in Regulatory Guide 1.52 (Ref. 4).~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

SR 3.7.~~13~~.4 3
12

This SR verifies the integrity of the ECCS pump room enclosure. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of the ECCS ~~PREACS~~. During the post accident mode of operation, the ECCS ~~PREACS~~ is designed to maintain a slight negative pressure in the ECCS pump room with respect to adjacent areas to prevent unfiltered LEAKAGE. The ECCS ~~PREACS~~ is designed to maintain this negative pressure at a flow rate of ≤ ~~[20,000]~~ 27,000 cfm from the ECCS pump room. ~~[The Frequency of [18] months is consistent with the guidance provided in the NUREG-0800, Section 6.5.1 (Ref. 8).~~

Area Ventilation System 1
Area Ventilation System 1
Area Ventilation System 2

~~This test is conducted with the tests for filter penetration; thus, an [18] month Frequency, on a STAGGERED TEST BASIS is consistent with other filtration SRs.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

~~[SR 3.7.13.5~~ 1 4

~~Operating the ECCS PREACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the bypass damper is verified if it can be opened. The SR excludes automatic dampers that are locked, sealed, or otherwise secured in the open position. The SR does not apply to dampers that are locked, sealed, or otherwise secured in the open position since the affected dampers were verified to be in the open position prior to being locked, sealed, or otherwise secured. Placing an automatic damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the damper to be closed to support the accident analysis. Restoration of an automatic damper to the closed position requires verification that the SR has been met within its required Frequency. [An [18] month Frequency is consistent with that specified in Reference 4.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]]~~

REFERENCES

1. FSAR, Section [~~6.5.1~~]. 6.3.1 U 1 2
2. FSAR, Section ~~[9.4.5]~~. 9.4.3 U 1 2
3. FSAR, Section ~~[15.6.5]~~. 15.4.1 U 1 2

BASES

REFERENCES (continued)

~~4. Regulatory Guide 1.52, Rev. [2].~~ 1

4 ~~5~~. 10 CFR ~~100.11.~~ 50.67 1

5 ~~6~~. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010. 1

6 ~~7~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001. 1

~~8. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.~~ 1

B 3.7 PLANT SYSTEMS

B 3.7.~~13~~ 12 Emergency Core Cooling System (ECCS) ~~Pump Room Exhaust Air Cleanup System (PREACS)~~

BASES

BACKGROUND

The ECCS ~~PREACS~~ filters air from the area of the active ECCS components during the recirculation phase of a loss of coolant accident (LOCA). The ECCS ~~PREACS~~, in conjunction with other, normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area and the lower reaches of the Auxiliary Building. R1

The ECCS ~~PREACS~~ consists of two independent and redundant trains. Each train consists of ~~a heater,~~ a prefilter ~~or demister~~, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system ~~, as well as demisters functioning to reduce the relative humidity of the air stream~~. ~~A~~ second bank ~~of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines and to back up the upstream HEPA filter, should it develop a leak~~. The system initiates filtered ventilation of the pump room and lower region of the Auxiliary Building following receipt of a safety injection actuation signal ~~or coolant injection actuation signal~~.

The ECCS ~~PREACS~~ is a standby system, parts of which may also operate during normal unit operations. The Reactor Auxiliary Building Main Ventilation System provides normal cooling. During emergency operations, the ECCS ~~PREACS~~ dampers are realigned and fans are started to initiate filtration. Upon receipt of ~~the actuating Engineered Safety Feature Actuation System~~ signal~~(s)~~, normal air discharges from the ECCS pump room, the pump room is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters ~~or demisters~~ remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ECCS ~~PREACS~~ is discussed in the FSAR, Sections ~~[~~6.5.1~~]~~, ~~[~~9.4.5~~]~~, and ~~[~~15.6.5~~]~~ (Refs. 1, 2, and 3, respectively), as it may be used for normal, as well as post accident, atmospheric cleanup functions. ~~The primary purpose of the heaters is to maintain the relative humidity at an acceptable level consistent with iodine removal efficiencies, as discussed in the Regulatory Guide 1.52 (Ref. 4).~~

BASES

Area Ventilation System

APPLICABLE SAFETY ANALYSES

The design basis of the ECCS ~~PREACS~~ is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as safety injection pump seal failure, during the recirculation mode. In such a case, the system limits the radioactive release to within ~~10 CFR 100~~ limits (Ref. ~~5~~); ~~or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits)~~. The analysis of the effects and consequences of a large break LOCA is presented in Reference 3. The ECCS ~~PREACS~~ also actuates following a small break LOCA, requiring the unit to go into the recirculation mode of long term cooling and to clean up releases of smaller leaks, such as from valve stem packing.

10 CFR 50.67

4

Area Ventilation System

1

The two types of system failures that are considered in the accident analysis are complete loss of function and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

Area Ventilation System

The ECCS ~~PREACS~~ satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

area ventilation

LCO

Two independent and redundant ECCS ~~PREACS~~ trains are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the ECCS pump room exceeding the required limits in the event of a Design Basis Accident (DBA).

Area Ventilation System

ECCS ~~PREACS~~ is considered OPERABLE when the individual components necessary to maintain the ECCS Pump Room filtration are OPERABLE in both trains.

area ventilation

An ECCS ~~PREACS~~ train is considered OPERABLE when its associated:

a. Fan is OPERABLE,

b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions, and

c. ~~Heater, demister,~~ ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

BASES

LCO (continued)

The LCO is modified by a Note allowing the ECCS pump room boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for ECCS pump room isolation is indicated.

APPLICABILITY

Area Ventilation System

In MODES 1, 2, 3, and 4, the ECCS ~~PREACS~~ is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS. 1

Area Ventilation System

In MODES 5 and 6, the ECCS ~~PREACS~~ is not required to be OPERABLE, since the ECCS is not required to be OPERABLE. 1

ACTIONS

A.1

area ventilation

Area Ventilation System

With one ECCS ~~PREACS~~ train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS ~~PREACS~~ function. 1 1

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time) and this system is not a direct support system for the ECCS. The 7 day Completion Time is reasonable, based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.

B.1

~~REVIEWER'S NOTE~~

~~Adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of an intentional and unintentional entry into Condition B.~~ 4

BASES

ACTIONS (continued)

area ventilation

If the ECCS pump room boundary is inoperable, the ECCS ~~PREACS~~ trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ECCS pump room boundary within 24 hours. During the period that the ECCS pump room boundary is inoperable, appropriate compensatory measures ~~[~~consistent with the intent, as applicable, of GDC 19~~, 60, 64 and 10 CFR Part 100]~~ should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ECCS pump room boundary.

area ventilation system

~~REVIEWER'S NOTE~~

~~Condition C is not applicable to plant designs that rely on the ECCS PREACS to provide temperature control of the ECCS pump rooms. Reference 6 did not evaluate the effect of two inoperable ECCS PREACS trains on the ECCS pump room cooling function.~~

C.1

area ventilation

With two ECCS ~~PREACS~~ trains inoperable for reasons other than an inoperable boundary, action must be taken to restore at least one ECCS ~~PREACS~~ train to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second ECCS ~~PREACS~~ train is intentionally declared inoperable. The Required Action is not intended for voluntary removal of redundant systems or components from service. The Required Action is only applicable if one train is inoperable for any reason and the second train is found to be inoperable, or if both trains are found to be inoperable at the same time. The Completion Time is based on Reference ~~6~~ which demonstrated that the 24 hour Completion Time is acceptable based on the low frequency of the potential challenge and the small incremental risk associated with continued operation.

area ventilation

5

BASES

ACTIONS (continued)

D.1 and D.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

2

area ventilation

1

If the ECCS ~~PREACS~~ train or ECCS pump room boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be in a MODE in which overall plant risk is minimized. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

6

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~7~~). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

1

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.~~13~~.1

12

periodically with initiation from the control room

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train ~~once a month~~ provides an adequate check on this system. Operation ~~[with the heaters on]~~ for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that ~~[heater failure,]~~ blockage, fan or motor failure, or excessive vibration can be detected for corrective action. ~~[The 31 day Frequency is based on the known reliability of equipment, and the two train redundancy available.~~

3
1
2
2
2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

2

SR 3.7.~~13~~.2

12

Area Ventilation System

This SR verifies that the required ECCS ~~PREACS~~ testing is performed in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. The ~~[~~VFTP~~]~~ includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~.

3
1
2

SR 3.7.~~13~~.3

12

area ventilation

This SR verifies that each ECCS ~~PREACS~~ train starts and operates on an actual or simulated actuation signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position

3
1

BASES

SURVEILLANCE REQUIREMENTS (continued)

prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. ~~[The [18] month Frequency is consistent with that specified in Regulatory Guide 1.52 (Ref. 4).~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

SR 3.7.~~13~~.4 12 3

This SR verifies the integrity of the ECCS pump room enclosure. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of the ECCS ~~PREACS~~. During the post accident mode of operation, the ECCS ~~PREACS~~ is designed to maintain a slight negative pressure in the ECCS pump room with respect to adjacent areas to prevent unfiltered LEAKAGE. The ECCS ~~PREACS~~ is designed to maintain this negative pressure at a flow rate of ≤ ~~[20,000]~~ 27,000 cfm from the ECCS pump room. ~~[The Frequency of [18] months is consistent with the guidance provided in the NUREG-0800, Section 6.5.1 (Ref. 8).~~

Area Ventilation System 1
Area Ventilation System 1
Area Ventilation System 2

~~This test is conducted with the tests for filter penetration; thus, an [18] month Frequency, on a STAGGERED TEST BASIS is consistent with other filtration SRs.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~[SR 3.7.13.5~~

~~Operating the ECCS PREACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the bypass damper is verified if it can be opened. The SR excludes automatic dampers that are locked, sealed, or otherwise secured in the open position. The SR does not apply to dampers that are locked, sealed, or otherwise secured in the open position since the affected dampers were verified to be in the open position prior to being locked, sealed, or otherwise secured. Placing an automatic damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the damper to be closed to support the accident analysis. Restoration of an automatic damper to the closed position requires verification that the SR has been met within its required Frequency. [An [18] month Frequency is consistent with that specified in Reference 4.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]]~~

REFERENCES

1. FSAR, Section ~~[~~6.5.1~~]~~.
2. FSAR, Section ~~[9.4.5]~~. 9.4.3
3. FSAR, Section ~~[~~15.6.5~~]~~.

BASES

REFERENCES (continued)

~~4. Regulatory Guide 1.52, Rev. [2].~~ 1

4 ► ~~5~~. 10 CFR ~~100.11.~~ ◄ 50.67 1

5 ► ~~6~~. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010. 1

6 ► ~~7~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001. 1

~~8. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.~~ 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.12, BASES, ECCS AREA VENTILATION SYSTEM

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS Specification number and title.

4. As stated in the Unit 1 and Unit 2 UFSAR, Section 9.4.3.1, the ECCS Area Ventilation System is sized to maintain a slightly negative pressure in the ECCS area with respect to surrounding areas of the reactor auxiliary building. Access into the ECCS pump room area from other areas of the reactor auxiliary building is through gasketed self-closing or locked closed doors. Opening of locked doors is controlled under administrative controls.

5. PSL Unit 1 and Unit 2 ECCS Area Ventilation System design does not rely upon the ECCS Area Ventilation System to provide temperature control of the ECCS pump rooms during normal operation. Under normal operation, the reactor auxiliary building main ventilation supply (HVS-4A/B) and exhaust system (HVE-10A/B) provides the necessary ventilation of the ECCS pump rooms. Under accident conditions when ECCS pumps are operating, the ventilation supply to the nonessential section of the reactor auxiliary building is directed to the pump rooms to provide the additional cooling air requirement. Dampers are positioned automatically on SIAS signal to provide the proper flow path for supply air to the ECCS pump areas. The ECCS Area Ventilation System fans (HVE-9A and 9B) are automatically energized and dampers in the exhaust ductwork are automatically positioned to allow the fans to draw all exhaust air from the ECCS pump areas through the HEPA and charcoal filter banks before exhausting the air to the outside atmosphere. Additionally, the ECCS Area Ventilation System establishes and maintains a slightly negative pressure relative to surrounding areas assuming leakage in the ECCS areas.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.12, ECCS AREA VENTILATION SYSTEM

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 13

3.7.13, Spent Fuel Storage Pool Water Level

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REFUELING OPERATIONS

SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

~~3.9.11~~ ~~The Spent Fuel Pool shall be maintained with:~~

spent

shall be ≥

LCO 3.7.13

~~a.~~ The fuel storage pool water level ~~greater than or equal to~~ 23 ft over the top of irradiated fuel assemblies seated in the storage racks~~, and~~

b. The fuel storage pool boron concentration greater than or equal to 1900 ppm. See ITS 3.7.14

Applicability

APPLICABILITY: ~~Whenever~~ irradiated fuel assemblies in the spent fuel storage pool.

During movement of

L01

ACTION:

Spent fuel storage pool

within limit

spent

ACTION A

~~a.~~ ~~With the~~ water level ~~requirement~~ not ~~satisfied~~, immediately suspend ~~all~~ movement of fuel assemblies ~~and crane operations with loads~~ in the fuel storage ~~areas and restore the water level to within its limit within 4 hours~~.

L02

L01

pool

irradiated

b. With the boron concentration requirement not satisfied, immediately suspend all movement of fuel assemblies in the fuel storage pool and initiate action to restore the fuel storage pool boron concentration to within the required limit. See ITS 3.7.14

ACTION A.1 Note

~~c.~~ The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

water level is ≥ 23 feet above the top of irradiated fuel assemblies seated in the storage racks

Verify

SR 3.7.13.1

~~4.9.11~~ ~~The water level in~~ the spent fuel storage pool ~~shall be determined to be at least its minimum required depth~~ in accordance with the Surveillance Frequency Control Program ~~when irradiated fuel assemblies are in the fuel storage pool~~.

L01

4.9.11.1 Verify the fuel storage pool boron concentration is within limit in accordance with the Surveillance Frequency Control Program. See ITS 3.7.14

A01

REFUELING OPERATIONS

3/4.9.11 SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

~~3.9.11~~ ~~The Spent Fuel Pool shall be maintained with:~~

spent

shall be ≥

LCO 3.7.13

a. The fuel storage pool water level ~~greater than or equal to~~ 23 ft over the top of irradiated fuel assemblies seated in the storage racks~~, and~~

b. The fuel storage pool boron concentration greater than or equal to 1900 ppm. — See ITS 3.7.14

Applicability

APPLICABILITY: ~~Whenever~~ irradiated fuel assemblies ~~are~~ in the spent fuel storage pool.

During movement of

L01

ACTION:

Spent fuel storage pool

within limit

spent

ACTION A

~~a.~~ ~~With the~~ water level ~~requirement~~ not ~~satisfied~~, immediately suspend ~~all~~ movement of fuel assemblies ~~and crane operations with loads~~ in the fuel storage ~~areas and restore the water level to within its limit within 4 hours~~.

irradiated

pool

L02

L01

b. With the boron concentration requirement not satisfied, immediately suspend all movement of fuel assemblies in the fuel storage pool and initiate action to restore fuel storage pool boron concentration to within the required limit. — See ITS 3.7.14

ACTION A.1 Note

~~c.~~ The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

water level is ≥ 23 feet above the top of irradiated fuel assemblies seated in the storage racks

Verify

SR 3.7.13.1

~~4.9.11~~ ~~The water level in~~ the spent fuel storage pool ~~shall be determined to be at least its minimum required depth~~ in accordance with the Surveillance Frequency Control Program ~~when irradiated fuel assemblies are in the fuel storage pool~~.

L01

4.9.11.1 Verify the fuel storage pool boron concentration is within limit in accordance with the Surveillance Frequency Control Program. — See ITS 3.7.14

A01

~~**INTENTIONALLY DELETED**~~

DISCUSSION OF CHANGES
ITS 3.7.13, SPENT FUEL STORAGE POOL WATER LEVEL

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.9.11 states that the requirements on spent fuel storage pool water level are applicable "Whenever irradiated fuel assemblies are in the spent fuel storage pool." CTS 4.9.11 requires the water level in the storage pool to be verified in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the spent fuel storage pool. ITS 3.7.13 is applicable "During movement of irradiated fuel assemblies in the spent fuel storage pool." ITS SR 3.7.14.1 requires verification of the spent fuel pool water level in accordance with the Surveillance Frequency Control Program. This changes the CTS by restricting the Applicability of the spent fuel pool water level Specification and performance of the Surveillance to only when there is a potential for a fuel handling accident, that is, during the movement of irradiated fuel assemblies in the spent fuel storage pool. In addition, since the Applicability is now limited to when irradiated fuel is being moved, the portion of CTS Action a. to restore water level to within its limit within 4 hours, after movement of fuel has been suspended, has also been deleted.

The purpose of CTS 3.9.11 is to ensure that the minimum spent fuel storage pool water level assumption in the fuel handling accident analysis is met. This change is acceptable because the requirements continue to ensure that the structures, systems, and components are maintained in the specified conditions assumed in

the safety analyses and licensing basis. According to UFSAR Section 15.4.3 (Unit 1) 15.7.4.1.2 (Unit 2), a fuel handling accident is initiated by the dropping of an irradiated fuel assembly either in the containment or in the fuel building and assumes all the fuel rods in the assembly are damaged. The PSL fuel handling accident analysis (outside containment) assumes a minimum water level of 23 feet above the damaged fuel assembly for both the containment and fuel handling building release locations. A minimum water level of 23 feet ensures an effective iodine decontamination factor of 200 per the guidance provided in NRC Regulatory Issue Summary 2006-04. A fuel handling accident is assumed to occur when an irradiated fuel assembly is being moved and is precluded when irradiated fuel assemblies are not being moved. Imposing the minimum spent fuel storage pool water level requirement whenever irradiated fuel assemblies are in the spent fuel storage pool is unnecessary. Therefore, the ITS imposes the controls on minimum spent fuel storage pool water level during the movement of irradiated fuel assemblies in the spent fuel storage pool. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.9.11 Action a. states that when the spent fuel pool water level is not met, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas. ITS 3.7.13 Required Action A.1 states that when spent fuel storage pool water level is not within limit, immediately suspend movement of irradiated fuel assemblies in the spent fuel storage pool. This changes the CTS by deleting the requirement to suspend crane operation over the spent fuel storage areas.

The purpose of the CTS 3.9.11 Action is to preclude a fuel handling accident from occurring when the initial conditions for that accident are not met. This change is acceptable because the Required Action places the plant in a condition where a fuel handling accident is precluded. The initiator to a fuel handling accident assumed in the accident analysis is the damaging of a single irradiated fuel assembly. Damaging a fuel assembly which has not been irradiated has no significant radiological effects and is not assumed in the fuel handling accident analysis. Therefore, stopping the handling of fuel assemblies which have not been irradiated when the spent fuel pool water level is less than the limit is not required.

Crane travel related requirements were relocated from the CTS in Unit 1 and Unit 2 PSL Amendments 190 and 134, respectively, (“St. Lucie Units 1 and 2 – Issuance of Amendments Regarding the Relocation of Spent Fuel Pool Crane Technical Specification Requirements (TAC NOS. MB5667 and MB5668),” dated April 28, 2004 (ADAMS Accession No. ML 040440111)) to the respective unit Updated Final Safety Analysis Report where the operational detail is controlled under 10 CFR 50.59 criteria. As cited in the NRC Safety Evaluation associated with Amendments 190 (Unit 1) and 134 (Unit 2), NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” regulatory guidelines for control of heavy load lifts provide assurance of safe handling of heavy loads in areas where a load drop could impact stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. Section 5.1.1 of NUREG 0612 provides guidelines for reducing the likelihood of dropping heavy loads and provides criteria for establishing safe load

paths; procedures for load-handling operations; training of crane operators; design, testing, inspection, and maintenance of cranes and lifting devices; and analyses of the impact of heavy load drops. The guidelines in Sections 5.1.2 through 5.1.6 address alternatives to either further reduce the probability of a load-handling accident or mitigate the consequences of heavy load drops. These alternatives include using a single-failure-proof crane to improve reliability through increased factors of safety and through redundancy or duality in certain active components. Criteria for design of single-failure-proof cranes are included in NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants.”

This change is acceptable because the possibility of damage to an irradiated fuel assembly as a consequence of mishandling components other than an irradiated fuel assembly is minimized by thorough training, detailed procedures and equipment design. The PSL crane design precludes the handling of heavy objects, such as shipping casks, over the spent fuel pool storage racks. Administrative controls prevent the movement of heavy loads over the cask pit whenever the cask pit rack is installed in the cask area of the spent fuel storage pool. In addition, the cask handling crane design meets the regulatory guidance for single-failure-proof cranes in NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants” and NUREG-0612. Administrative controls that control the movement of light loads or prevent movement of light loads over irradiated fuel assemblies are similar to those used for control of heavy loads, to the extent practicable, as advised in NUREG 0612. Consequently, the possibility of dropping a load other than an irradiated fuel assembly and damaging of irradiated fuel assemblies in the spent fuel storage pool is remote. Therefore, the CTS 3.9.11 action related to suspension of crane operation with loads in the fuel storage areas is not necessary to be included in the technical specifications and is removed.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

Spent
3.7.~~16~~ 13 Fuel Storage Pool Water Level

3.9.11.a LCO 3.7.~~16~~ 13 The spent fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

Applicability APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a. Action c. Spent A. Fuel storage pool water level not within limit.	A.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ---------- Suspend movement of irradiated fuel assemblies in spent fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.9.11 SR 3.7.~~16~~ 13.1 Verify the spent fuel storage pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

3.7 PLANT SYSTEMS

Spent
3.7.~~16~~ 13 Fuel Storage Pool Water Level

3.9.11.a LCO 3.7.~~16~~ 13 The spent fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

Applicability APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a. Action c. Spent A. Fuel storage pool water level not within limit.	A.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ---------- Suspend movement of irradiated fuel assemblies in spent fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.9.11 SR 3.7.~~16~~ 13.1 Verify the spent fuel storage pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.13, SPENT FUEL STORAGE POOL WATER LEVEL

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS Specification number and title. The numbering is changed from 3.7.16 to 3.7.13, and the title is changed to “Spent Fuel Pool Water Level.”

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.~~16~~ Spent 13 Fuel Storage Pool Water Level

Sufficient water is necessary to retain iodine fission product activity in the water in the event of an FHA (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to within the radiation dose criteria set forth in 10 CFR 50.67, as provided by the guidance of Reference 1.

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident (FHA). The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the FSAR, Section ~~[~~9.1.2~~]~~, Reference ~~1~~, and the Spent Fuel Pool Cooling and ~~Cleanup~~ Purification System is given in the FSAR, Section ~~[~~9.1.3~~]~~ (Ref. ~~2~~). The assumptions of the fuel handling accident are given in the FSAR, Section ~~[15.7.4]~~ 15.4.3 (Ref. ~~3~~ 2).

APPLICABLE SAFETY ANALYSES

~~The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits.~~

~~According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With a 23 ft water level, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle, dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the bundle and the surface, by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rods fail from a hypothetical maximum drop.~~ INSERT 1

The fuel storage pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The specified water level preserves the assumptions of the ~~fuel handling accident~~ FHA analysis (Ref. ~~3~~ 2). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool since the potential for a release of fission products exists.

1

INSERT 1

The FHA analysis is described in Reference 2. During movement of irradiated fuel assemblies, the water level in the spent fuel storage pool is an initial condition design parameter in the analysis of the FHA in the spent fuel storage pool postulated in Appendix B of Regulatory Guide (RG) 1.183 (Ref. 1). A minimum water level of 23 feet is maintained above the damaged fuel assembly for both the containment and fuel handling building (FHB) release locations. Fuel handling accidents consist of dropping a single irradiated fuel assembly either in the FHB or inside the containment and assumes all the fuel rods in the assembly are damaged. The offsite dose analysis does not credit filtration during an FHA inside containment with the maintenance hatch open or in the spent fuel storage pool. The control room dose analysis credits control room emergency ventilation filtration during an FHA inside or outside the containment.

In the event of an FHA, a minimum water level of 23 feet and a minimum decay time of 72 hours prior to fuel handling ensures an effective iodine decontamination factor of 200 per the guidance provided in NRC Regulatory Issue Summary 2006-04 (Ref. 5). The FHA assumes 100% of the noble gas is released from the damaged fuel assembly and escapes from the pool. All of the non-iodine particulates released from the damaged fuel assembly are assumed to be retained by the water. Iodine released from the damaged fuel assembly is assumed to be composed of 99.85% elemental and 0.15% organic. Activity released from the FHA is assumed to leak to the environment over a two-hour period and no credit is taken for dilution in the spent fuel pool level.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended. This ~~effectively~~ precludes ~~a spent fuel handling accident~~ an FHA from occurring. This does not preclude moving a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~16~~13.1

This SR verifies sufficient fuel storage pool water is available in the event of ~~a fuel handling accident~~ an FHA. The water level in the fuel storage pool must be checked periodically. ~~[The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~During refueling operations, the level in the fuel storage pool is at equilibrium with that of the refueling canal, and the level in the refueling canal is checked daily in accordance with LCO 3.7.17, "Fuel Storage Pool Boron Concentration."~~

BASES

REFERENCES

3 → ~~1~~. FSAR, Section ~~[~~9.1.2~~]~~.
U
4 → ~~2~~. FSAR, Section ~~[~~9.1.3~~]~~.
U
2 → ~~3~~. FSAR, Section ~~[15.7.4]~~. ← 15.4.3
U
1 → ~~4~~. Regulatory Guide ~~1.25~~. ← 1.183, July 2000

5. ~~10 CFR 100.11.~~ ← NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternate Source Terms," March 7, 2006.

2
2
2
1
1

B 3.7 PLANT SYSTEMS

B 3.7.~~16~~ Spent 13 Fuel Storage Pool Water Level

Sufficient water is necessary to retain iodine fission product activity in the water in the event of an FHA (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to within the radiation dose criteria set forth in 10 CFR 50.67, as provided by the guidance of Reference 1.

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident (FHA). The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the FSAR, Section ~~[~~9.1.2~~]~~, Reference ~~1~~, and the Spent Fuel Pool Cooling and ~~Cleanup~~ Purification System is given in the FSAR, Section ~~[~~9.1.3~~]~~ (Ref. ~~2~~). The assumptions of the fuel handling accident are given in the FSAR, Section ~~[~~15.7.4~~]~~ (Ref. ~~3~~).

APPLICABLE SAFETY ANALYSES

~~The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits.~~

INSERT 1

~~According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With a 23 ft water level, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle, dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the bundle and the surface, by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rods fail from a hypothetical maximum drop.~~

The fuel storage pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The specified water level preserves the assumptions of the ~~fuel handling accident~~ FHA analysis (Ref. ~~3~~). As such, it is the minimum required for fuel storage and movement within the fuel storage poo l.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool since the potential for a release of fission products exists.

1

INSERT 1

The FHA analysis is described in Reference 2. During movement of irradiated fuel assemblies, the water level in the spent fuel storage pool is an initial condition design parameter in the analysis of the FHA in the spent fuel storage pool postulated in Appendix B of Regulatory Guide (RG) 1.183 (Ref. 1). A minimum water level of 23 feet is maintained above the damaged fuel assembly for both the containment and fuel handling building (FHB) release locations. Fuel handling accidents consist of dropping a single irradiated fuel assembly either in the FHB or inside the containment and assumes all the fuel rods in the assembly are damaged. The offsite dose analysis does not credit filtration during an FHA inside containment with the maintenance hatch open or in the spent fuel storage pool. The control room dose analysis credits control room emergency ventilation filtration during an FHA inside or outside the containment.

In the event of an FHA, a minimum water level of 23 feet and a minimum decay time of 72 hours prior to fuel handling ensures an effective iodine decontamination factor of 200 per the guidance provided in NRC Regulatory Issue Summary 2006-04 (Ref. 5). The FHA assumes 100% of the noble gas is released from the damaged fuel assembly and escapes from the pool. All of the non-iodine particulates released from the damaged fuel assembly are assumed to be retained by the water. Iodine released from the damaged fuel assembly is assumed to be composed of 99.85% elemental and 0.15% organic. Activity released from the FHA is assumed to leak to the environment over a two-hour period and no credit is taken for dilution in the spent fuel pool level.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended. This ~~effectively~~ precludes ~~a spent fuel handling accident~~ an FHA from occurring. This does not preclude moving a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~16~~13.1

This SR verifies sufficient fuel storage pool water is available in the event of ~~a fuel handling accident~~ an FHA. The water level in the fuel storage pool must be checked periodically. ~~[The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~During refueling operations, the level in the fuel storage pool is at equilibrium with that of the refueling canal, and the level in the refueling canal is checked daily in accordance with LCO 3.7.17, "Fuel Storage Pool Boron Concentration."~~

BASES

REFERENCES

3 → ~~1~~. FSAR, Section ~~[~~9.1.2~~]~~. U

4 → ~~2~~. FSAR, Section ~~[~~9.1.3~~]~~. U

2 → ~~3~~. FSAR, Section ~~[~~15.7.4~~]~~. U

1 → ~~4~~. Regulatory Guide ~~1.25~~. ← 1.183, July 2000

5. ~~10 CFR 100.11.~~ ← NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternate Source Terms," March 7, 2006.

2 2 2 1 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.13, BASES, SPENT FUEL STORAGE POOL WATER LEVEL

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the ITS Specification number and title. The numbering is changed from 3.7.16 to 3.7.13, and the title is changed to “Spent Fuel Pool Water Level.”

4. ISTS SR 3.7.16.1 Bases refers to the incorrect ISTS LCO regarding refueling water level. Also, this statement does not provide any additional supporting information related to the Surveillance associated with spent fuel storage pool water level. Therefore, the statement is not included in the Bases for ITS SR 3.7.13.1

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.13, SPENT FUEL STORAGE POOL WATER LEVEL

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 14

3.7.14, Spent Fuel Storage Pool Boron Concentration

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REFUELING OPERATIONS

SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

~~3.9.11~~ ~~The Spent Fuel Pool shall be maintained with:~~

a. The fuel storage pool water level greater than or equal to 23 ft over the top of irradiated fuel assemblies seated in the storage racks, and

See ITS 3.7.13

spent

shall be ≥

LCO 3.7.14

~~b.~~ The fuel storage pool boron concentration ~~greater than or equal to~~ 1900 ppm.

stored

Applicability

APPLICABILITY: When~~ever~~ ~~irradiated~~ fuel assemblies are in the spent fuel storage pool.

Add proposed ITS 3.7.14 Applicability (second part) — L01

ACTION:

a. With the water level requirement not satisfied, immediately suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

See ITS 3.7.13

Spent fuel storage pool

within limit

Condition A

~~b.~~ ~~With the~~ boron concentration ~~requirement~~ not ~~satisfied~~, immediately suspend

ACTION A.1

~~all~~ movement of fuel assemblies in the fuel storage pool and initiate action to

ACTION A.2.1

restore ~~the~~ fuel storage pool boron concentration to within ~~the required~~ limit.

spent

spent

Add proposed ITS 3.7.14 ACTION A.2.2 — L01

ACTION A.1, A.2.1, and A.2.2 Note

~~c.~~ The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel storage pool shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the fuel storage pool.

See ITS 3.7.13

spent

SR 3.7.14.1

~~4.9.11.1~~ Verify the fuel storage pool boron concentration is within limit in accordance with the Surveillance Frequency Control Program.

A01

REFUELING OPERATIONS

3/4.9.11 SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

~~3.9.11~~ ~~The Spent Fuel Pool shall be maintained with:~~

a. The fuel storage pool water level greater than or equal to 23 ft over the top of irradiated fuel assemblies seated in the storage racks, and

See ITS 3.7.13

spent

shall be ≥

LCO 3.7.14

~~b.~~ The fuel storage pool boron concentration ~~greater than or equal to~~ 1900 ppm.

stored

Applicability

APPLICABILITY: When~~ever~~ ~~irradiated~~ fuel assemblies are in the spent fuel storage pool.

Add proposed ITS 3.7.14 Applicability (second part) — L01

ACTION:

a. With the water level requirement not satisfied, immediately suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

See ITS 3.7.13

Spent fuel storage pool

within limit

Condition A

~~b.~~ ~~With the~~ boron concentration ~~requirement~~ not ~~satisfied~~, immediately suspend

ACTION A.1

~~all~~ movement of fuel assemblies in the fuel storage pool and initiate action to

ACTION A.2.1

restore fuel storage pool boron concentration to within ~~the required~~ limit.

spent

spent

Add proposed ITS 3.7.14 ACTION A.2.2 — L01

ACTION A.1, A.2.1, and A.2.2 Note

~~c.~~ The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel storage pool shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the fuel storage pool.

See ITS 3.7.13

spent

~~4.9.11.1~~ Verify the fuel storage pool boron concentration is within limit in accordance with the Surveillance Frequency Control Program.

DISCUSSION OF CHANGES
ITS 3.7.14, SPENT FUEL STORAGE POOL BORON CONCENTATIION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.9.11 is applicable whenever irradiated fuel assemblies are in the spent fuel storage pool. ITS 3.7.14 is applicable when fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool. In addition, ITS 3.7.14 Required Action A.2.2 provides an alternative action to allow exiting the specified condition in the Applicability when the LCO is not met. This changes the CTS by adding the Applicability statement “and a spent fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool,” and adding an ACTION that allows exiting the Applicability if the LCO is not met.

The purpose of CTS 3.9.11 is to ensure adequate dissolved boron is in the spent fuel storage pool water to maintain the required subcriticality margin. This change is acceptable because the requirements continue to ensure that the process variable is maintained in the specified condition assumed in the safety analyses and licensing basis. Following performance of a fuel storage pool verification to ensure the fuel assemblies are in the prescribed spent fuel storage rack location, there is no potential for criticality based on the design of the storage racks. Performing a fuel storage pool verification provides assurance that no fuel assemblies have been inadvertently misplaced in the fuel storage pool. This change is consistent with the ISTS and is designated as less

restrictive because the LCO requirements are applicable in a less stringent operating condition than in the CTS and because less stringent Required Actions are being applied in the ITS than applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7.~~17~~ 14 Spent Fuel Storage Pool Boron Concentration

3.9.11.b LCO 3.7.~~17~~ 14 The spent fuel storage pool boron concentration shall be ≥ ~~[2000]~~ 1900 ppm.

Applicability L01 APPLICABILITY: When fuel assemblies are stored in the spent fuel storage pool and a spent fuel storage pool verification has not been performed since the last movement of fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b. Action c. A. Spent Fuel storage pool boron concentration not within limit.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
Action b.	A.1 Suspend movement of fuel assemblies in the spent fuel storage pool.	Immediately
	AND	
Action b.	A.2.1 Initiate action to restore spent fuel storage pool boron concentration to within limit.	Immediately
	OR	
L01	A.2.2 Initiate action to perform a spent fuel storage pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1 (14)	Verify the spent fuel storage pool boron concentration is within limit.	[7 days OR In accordance with the Surveillance Frequency Control Program]

4.9.11.1

1 2

3.7 PLANT SYSTEMS

3.7.~~17~~ 14 Spent Fuel Storage Pool Boron Concentration

3.9.11.b LCO 3.7.~~17~~ 14 The spent fuel storage pool boron concentration shall be ≥ ~~[2000]~~ 1900 ppm.

Applicability L01 APPLICABILITY: When spent fuel assemblies are stored in the spent fuel storage pool and a spent fuel storage pool verification has not been performed since the last movement of spent fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b. Action c. A. Spent Fuel storage pool boron concentration not within limit.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
Action b.	A.1 Suspend movement of spent fuel assemblies in the spent fuel storage pool.	Immediately
	AND	
Action b.	A.2.1 Initiate action to restore spent fuel storage pool boron concentration to within limit.	Immediately
	OR	
L01	A.2.2 Initiate action to perform a spent fuel storage pool verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1 (14)	Verify the spent fuel storage pool boron concentration is within limit.	[7 days OR In accordance with the Surveillance Frequency Control Program]

4.9.11.1

1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.14, SPENT FUEL STORAGE POOL BORON CONCENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.~~17~~ Spent 14 Fuel Storage Pool Boron Concentration 1

assuming a minimum soluble boron concentration in the spent fuel storage pool of 500 ppm for normal conditions and 1500 ppm for accident conditions

separate and distinct regions, each with various prescribed patterns,

BASES

15 Pool

BACKGROUND

As described in LCO 3.7.~~18~~, "Spent Fuel ~~Assembly~~ Storage," fuel assemblies are stored in the spent fuel racks ~~[~~in ~~a "checkerboard" pattern]~~ in accordance with criteria based on ~~[~~initial enrichment and discharge burnup~~]~~. Although the water in the spent fuel pool is normally borated to ≥ ~~[1800]~~ ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed ~~without taking credit for boron~~. 1

1900 storage 2 2 1

15

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the activity of the spent fuel pool storage regions. However, accidents can be postulated that could increase the reactivity.

A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.~~18~~ (e.g., an unirradiated fuel assembly ~~or an insufficiently depleted fuel assembly~~). ~~This accident is analyzed assuming the extreme case of completely loading the fuel pool racks with unirradiated assemblies of maximum enrichment.~~ Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. 1

The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The fuel storage pool boron concentration is required to be ≥ 1900 ppm.

The specified concentration of dissolved boron in the fuel pool preserves the assumptions used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel pool. 1

storage

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly. 1

BASES

ACTIONS A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit. Alternately, beginning a verification of the spent fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS SR 3.7.17.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. ~~[The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

REFERENCES None.

B 3.7 PLANT SYSTEMS

B 3.7.~~17~~ Spent 14 Fuel Storage Pool Boron Concentration

assuming a minimum soluble boron concentration in the spent fuel storage pool of 500 ppm for normal conditions and 1500 ppm for accident conditions

separate and distinct regions, each with various prescribed patterns,

BASES

BACKGROUND

As described in LCO 3.7.~~18~~ 15, "Spent Fuel ~~Assembly~~ Pool Storage," fuel assemblies are stored in the spent fuel racks ~~[~~in ~~a "checkerboard" pattern]~~ in accordance with criteria based on ~~[~~initial enrichment and discharge burnup~~]~~. Although the water in the spent fuel pool is normally borated to ≥ ~~[1800]~~ 1900 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed ~~without taking credit for boron~~.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the activity of the spent fuel pool storage regions. However, accidents can be postulated that could increase the reactivity.

A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.~~18~~ 15 (e.g., an unirradiated fuel assembly ~~or an insufficiently depleted fuel assembly~~). ~~This accident is analyzed assuming the extreme case of completely loading the fuel pool racks with unirradiated assemblies of maximum enrichment.~~ Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios.

The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The fuel storage pool boron concentration is required to be ≥ 1900 ppm.

The specified concentration of dissolved boron in the fuel pool preserves the assumptions used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

BASES

ACTIONS A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit. Alternately, beginning a verification of the spent fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS SR 3.7.~~17~~.1 (14)

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. ~~[The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

REFERENCES None.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.14, BASES, SPENT FUEL STORAGE POOL BORON CONCENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.14, SPENT FUEL STORAGE POOL BORON CONCENTATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 15

3.7.15, Spent Fuel Pool Storage

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

DESIGN FEATURES

CRITICALITY (Continued)

4. For storage of enriched fuel assemblies, requirements of Criteria in 5.6.1.a.1 and 5.6.1.a.3 shall be met by positioning fuel in the spent fuel storage racks consistent with the requirements of Specification 5.6.1.c.

5. Vessel Flux Reduction Assemblies (VFRAs), as defined in Section 9.1 of the Updated Final Safety Analysis Report, may be placed in any allowable fuel storage location.

6. Fissile material, not contained in a fuel assembly lattice, shall be stored in accordance with the requirements of Criteria in 5.6.1.a.1 and 5.6.1.a.3.

7. The Metamic neutron absorber inserts shall have a ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$.

See ITS 4.0

b. The Region 1 cask pit storage rack shall contain neutron absorbing material (Boral) between stored fuel assemblies when installed in the spent fuel pool.

LCO 3.7.15

Add proposed LCO 3.7.15 and Applicability

A02

c. ~~Loading of spent fuel storage racks shall be controlled as described below.~~ Criteria ~~in~~ ~~5.6.1.c.2, 5.6.1.c.3, 5.6.1.c.5 and 5.6.1.c.6~~ do not apply to the Region 1 cask pit storage rack.

b, c, e, and f

LCO 3.7.15.a

a. ~~1~~. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel storage rack shall be less than or equal to 4.6 weight percent.

LCO 3.7.15.b

b. ~~2~~. Fuel placed in Region 1 of the spent fuel pool storage racks shall comply with the storage patterns and alignment restrictions of Figure ~~5.6~~-1 and the minimum burnup requirements of Table ~~5.6~~-1.

3.7.15

3.7.15

3.7.15

LCO 3.7.15.c

c. ~~3~~. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage patterns or allowed special arrangements of Figure ~~5.6~~-2 and the minimum burnup requirements of Table ~~5.6~~-1. The allowed special arrangement for fresh fuel may be repeated, provided the applicable interface requirements specified by the safety analysis are met.

3.7.15

LCO 3.7.15.d

d. ~~4~~. Any fuel satisfying ~~criteria 5.6.1.c.1~~, including fresh fuel, may be placed in the Region 1 cask pit storage rack.

Specification 3.7.15.a

LCO 3.7.15.e

e. ~~5~~. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required.

LCO 3.7.15.f

f. ~~6~~. Any 2x2 array of Region 2 storage cells that interface with Region 1 shall comply with the rules of Figure ~~5.6~~-3. The allowed special arrangement in Region 2 as shown in Figure ~~5.6~~-2 shall not be placed adjacent to Region 1.

3.7.15

d. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a maximum planar average U-235 enrichment less than or equal to 4.6 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 under the most reactive condition.

See ITS 4.0

LCO 3.7.15 ACTION A

Add proposed LCO 3.7.15 ACTION A

A02

SR 3.7.15.1

Add proposed SR 3.7.15.1

M01

A01

~~TABLE 5.6-1~~ Table 3.7.15-1 (page 1 of 1)

Table 3.7.15-1

Minimum Burnup as a Function of Enrichment

Fuel Type	Cooling Time (Years)	Coefficients		
		A	B	C
1	0	-36.6860	22.4942	-1.4413
2	0	-36.1742	16.6000	-0.8958
3	0	-34.7091	23.1361	-1.6204
4	0	-24.5145	21.3404	-1.2444
	2.5	-26.8311	22.5246	-1.5029
	5	-24.7233	20.9763	-1.3246
	10	-23.6285	19.9541	-1.2505
	15	-23.5458	19.9336	-1.3180
	20	-22.4382	19.2460	-1.2629
5	0	-8.1856	14.5275	-0.0719
	2.5	-11.8506	16.1475	-0.3969
	5	-16.5196	18.5309	-0.7837
	10	-13.6831	16.3475	-0.5844
	15	-12.5819	15.6175	-0.5656
	20	-12.6469	15.4575	-0.5906

NOTES:

1. To qualify in a ”fuel type,” the burnup of a fuel assembly must exceed the minimum burnup “BU” calculated by inserting the “coefficients” for the associated “fuel type” and “cooling time” into the polynomial function:

   $BU = A + B*E + C*E^2$, where:

   BU = Minimum Burnup (GWD/MTU)

   E = Initial Maximum Planar Average Enrichment (weight percent uranium-235)

   A, B, C = Coefficients

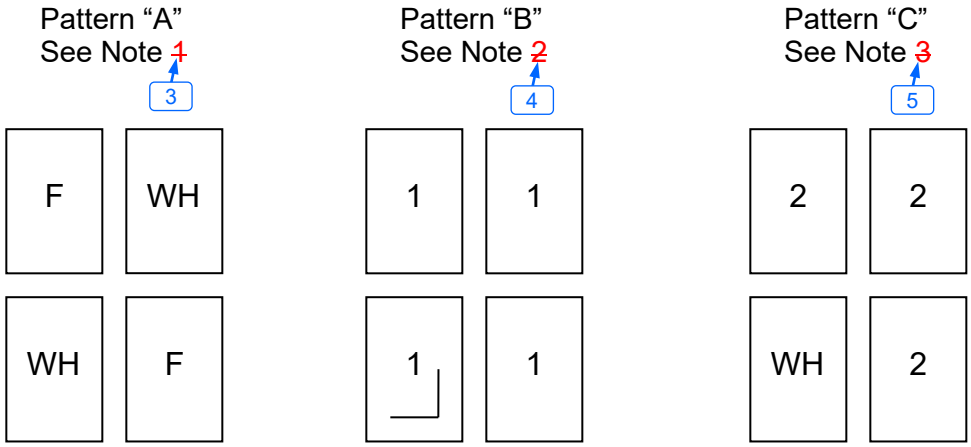
2. Interpolation between values of cooling time is not permitted.

A01

Figure 3.7.15-1 (page 1 of 1)

Allowable Checkerboard Storage Patterns
(See Notes ~~4~~ and ~~5~~)

1 2



NOTES:

3 → ~~1~~. F represents Fresh Fuel. WH represents an empty cell. Allowable Pattern is Fresh Fuel checkerboarded with empty cells. Diagram is for illustration only.

4 → ~~2~~. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 1 is defined in Table ~~5.6-1~~. Allowable pattern is at least one insert [either Metamic or full-length full-strength CEA] in any one of the 2x2 array locations. Diagram is for illustration only.

3.7.15-1

5 → ~~3~~. Numbering denotes fuel assembly type. WH represents an empty cell. Minimum burnup for fuel assembly type 2 is defined in Table ~~5.6-1~~. Allowable pattern is at least one empty cell in any of the 2x2 array locations. Diagram is for illustration only.

3.7.15-1

1 → ~~4~~. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2 → ~~5~~. Empty cells within any pattern are acceptable.

Figure 3.7.15-1 (page 1 of 1)

Figure 3.7.15-1 (page 1 of 1)

~~FIGURE 5.6-1~~

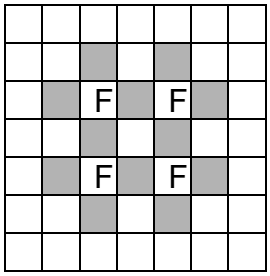
Allowable Region 1 Storage Patterns and Fuel Arrangements

A01

ALLOWED SPECIAL ARRANGEMENT

Figure 3.7.15-2 (page 1 of 2)

Fresh Fuel Assemblies in Pattern “C”, “D”, or “E” Racks



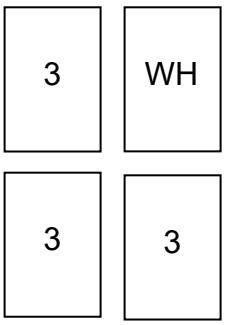
F = FRESH FUEL ASSEMBLY

= EMPTY CELL

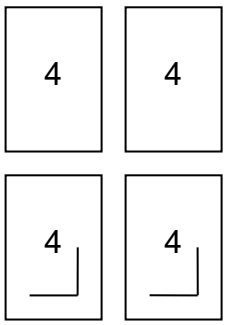
1 2

ALLOWABLE CHECKERBOARD STORAGE PATTERNS (See Notes ~~4~~ and ~~5~~)

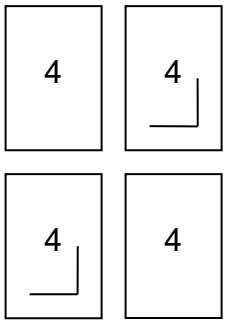
Pattern “C”
See Note ~~1~~
3



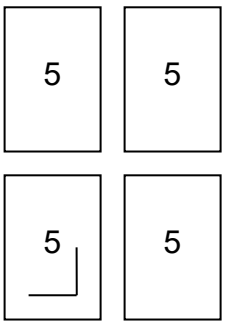
Pattern “D”
See Note ~~2~~
4



OR



Pattern “E”
See Note ~~3~~
5



OR

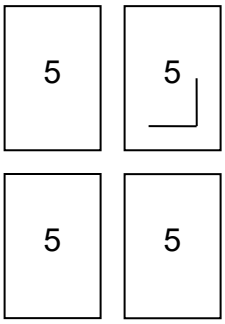


Figure 3.7.15-2 (page 1 of 2)

Figure 3.7.15-2 (page 1 of 2)

~~FIGURE 5.6-2 (Sheet 1 of 2)~~

Allowable Region 2 Storage Patterns and Fuel Alignments

A01

Figure 3.7.15-2 (page 2 of 2)

NOTES to ~~Figure 5.6-2~~

Figure 3.7.15-2 (page 1 of 2)

NOTES:

3.7.15-1

3 → ~~1~~. Numbering denotes fuel assembly type. WH represents an empty cell. Minimum burnup for fuel assembly type 3 is defined in Table ~~5.6-1~~. Allowable pattern is at least one empty cell in any of the 2x2 array locations. Diagram is for illustration only.

3.7.15-1

4 → ~~2~~. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 4 is defined in Table ~~5.6-1~~. Allowable pattern is at least two inserts, (either Metamic or full-length, full-strength CEA) in the 2x2 array. Diagrams are for illustration only.

3.7.15-1

5 → ~~3~~. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 5 is defined in Table ~~5.6-1~~. Allowable pattern is one insert, (either Metamic or full-length, full-strength CEA) in the 2x2 array. Diagrams are for illustration only.

1 → ~~4~~. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2 → ~~5~~. Empty cells within any pattern are acceptable.

Figure 3.7.15-2 (page 2 of 2)

Figure 3.7.15-2 (page 2 of 2)

~~FIGURE 5.6-2 (Sheet 2 of 2)~~

Allowable Region 2 Storage Patterns and Fuel Arrangements

A01

1 2

Allowed Region 2 to Region 1 Fuel Alignments
(See Notes ~~4~~ and ~~5~~)

Figure 3.7.15-3 (page 1 of 2)

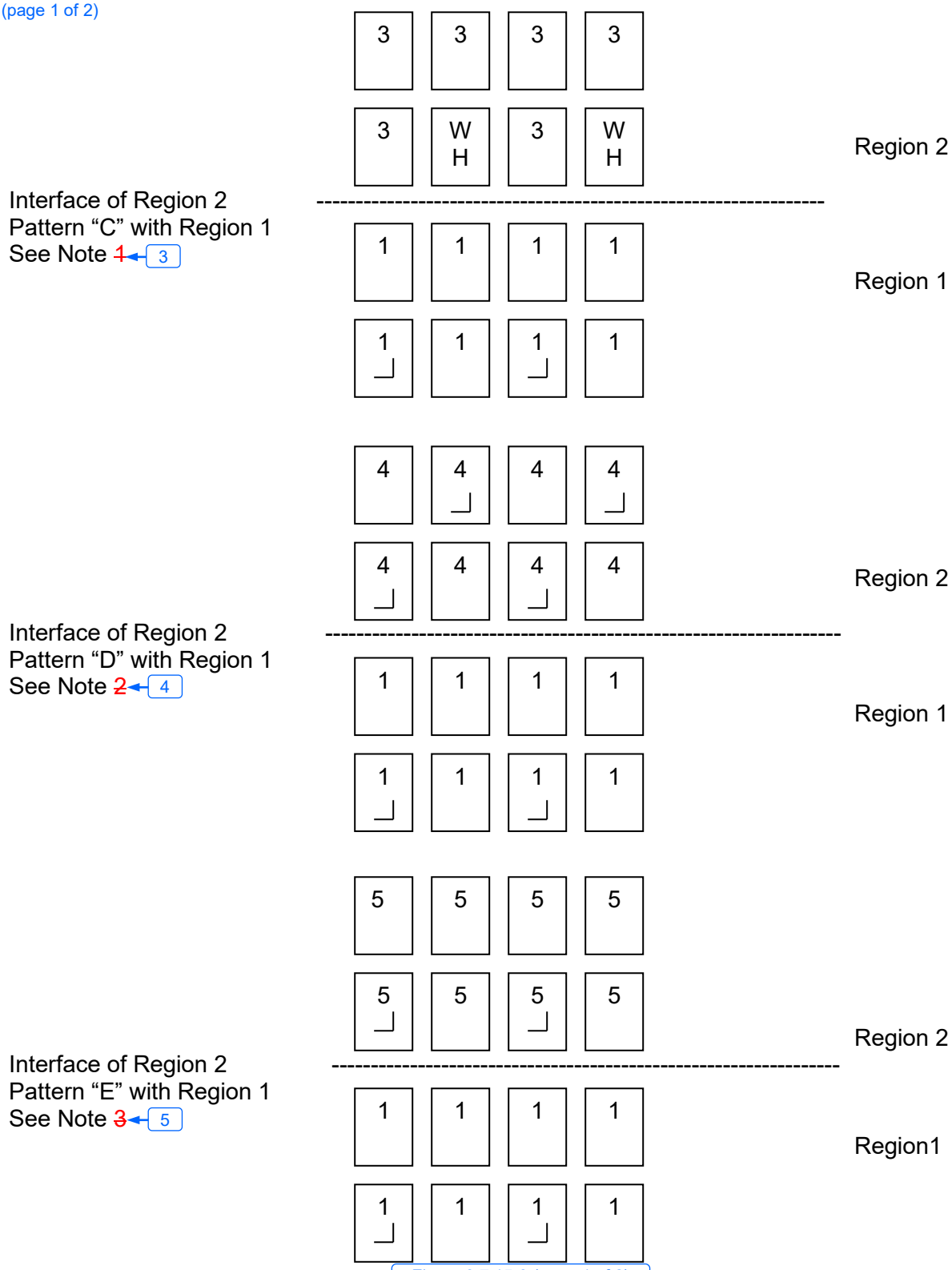


Figure 3.7.15-3 (page 1 of 2)

Figure 3.7.15-3 (page 1 of 2)

~~FIGURE 5.6-3 (Sheet 1 of 2)~~

Region 2 Interface Requirements with Region 1

A01

Figure 3.7.15-3 (page 2 of 2)

NOTES to ~~Figure 5.6-3~~

Figure 3.7.15-3 (page 1 of 2)

NOTES:

3 → ~~1~~. WH represents an empty cell. For the interface of Pattern "C" with Region 1, the empty cell must be on the rack periphery facing Region 1 racks. Diagrams are for illustration only.

4 → ~~2~~. For the interface of pattern "D" with Region 1, at least one cell on the rack periphery facing Region 1 rack must contain an insert (either Metamic of full-length full-strength CEA) in the 2x2 array. If the insert is Metamic, the insert must be oriented so that the corner of the L-shape is located closest to the Region 1 rack. Diagram is for illustration only.

5 → ~~3~~. For the interface of Pattern "E" with Region 1, the insert must be on the rack periphery facing the Region 1 rack. The insert may be either a Metamic of full-length full strength CEA. If the insert is Metamic, the insert must be oriented so that the corner of the L-shape is located closest to the Region 1 rack. Diagram is for illustration only.

1 → ~~4~~. Empty cells with any pattern are acceptable.

2 → ~~5~~. There are no interface requirements within Region 1. Any Pattern within Region 1 may be used for the interface. Pattern "B" was used only as an illustration.

Figure 3.7.15-3 (page 2 of 2)

Figure 3.7.15-3 (page 2 of 2)

~~FIGURE 5.6-3 (Sheet 2 of 2)~~

Region 2 Interface requirements with Region 1

A01

DESIGN FEATURES

5.5 DELETED

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

1. A k_{eff} equivalent to less than 1.0 when flooded with unborated water, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

See ITS 4.0

2. A k_{eff} equivalent to less than or equal to 0.95 when flooded with water containing 500 ppm boron, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

3. A nominal 8.965 inch center-to-center distance between fuel assemblies placed in the spent fuel pool storage racks and a nominal 8.80 inch center-to-center distance between fuel assemblies placed in the cask pit storage rack.

4. For storage of enriched fuel assemblies, requirements of Specification 5.6.1.a.1 and 5.6.1.a.2 shall be met by positioning fuel in the spent fuel pool storage racks consistent with the requirements of Specification 5.6.1.c.

5. Fissile material, not contained in a fuel assembly lattice, shall be stored in accordance with the requirements of Specifications 5.6.1.a.1 and 5.6.1.a.2.

6. The Metamic neutron absorber inserts shall have a ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$.

b. The cask pit storage rack shall contain neutron absorbing material (Boral) between stored fuel assemblies when installed in the spent fuel pool.

LCO 3.7.15

Add proposed LCO 3.7.15 and Applicability

A02

~~c. Loading of spent fuel pool storage racks shall be controlled as described below.~~

LCO 3.7.15.a

a → ~~1~~. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel pool storage rack shall be less than or equal to 4.6 weight percent.

LCO 3.7.15.b

b → ~~2~~. Fuel placed in Region 1 of the spent fuel pool storage racks shall comply with the storage pattern definitions of Figure ~~5.6~~-1 and the minimum burnup requirements as defined in Table ~~5.6~~-1. (See Specification ~~5.6.1.c.7~~ for exceptions)

3.7.15 3.7.15 3.7.15.g

LCO 3.7.15.c

c → ~~3~~. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions or allowed special arrangement definitions of Figure ~~5.6~~-2 and the minimum burnup requirements as defined in Table ~~5.6~~-1 (See Specification ~~5.6.1.c.7~~ for exceptions)

3.7.15 3.7.15.g

A01

~~DESIGN FEATURES~~ ~~(continued)~~

~~CRITICALITY~~ ~~(continued)~~

~~5.6.1 c. (continued)~~

LCO 3.7.15.d

d → ~~4~~. The 2x2 array of fuel assemblies that span the interface between Region 1 and Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions of Figure ~~5.6~~-3 and the minimum burnup requirements as defined in Table 5.6-1. The allowed special arrangements in Region 2 as shown in Figure ~~5.6~~-2 shall not be placed adjacent to Region 1. (See Specification ~~5.6.1.c.7~~ for exceptions)

3.7.15.g 3.7.15 3.7.15

LCO 3.7.15.e

e → ~~5~~. Fuel placed in the cask pit storage rack shall comply with the storage pattern definitions of Figure ~~5.6~~-4 and the minimum burnup requirements as defined in Table ~~5.6~~-1. (See Specification ~~5.6.1.c.7~~ for exceptions)

3.7.15 3.7.15.g

LCO 3.7.15.f

f → ~~6~~. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required.

LCO 3.7.15.g

g → ~~7~~. Fresh or spent fuel in any allowed configuration may be replaced with non-fuel hardware, and fresh fuel in any allowed configuration may be replaced with a fuel rod storage basket containing fuel rod(s). ~~Also,~~ storage of Metamic inserts or control rods, without any fissile material, is acceptable in locations designated as completely water-filled cells.

d. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a maximum planar average U-235 enrichment less than or equal to 4.6 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 under the most reactive condition.

See ITS 4.0

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

CAPACITY

5.6.3 The spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1491 fuel assemblies, and the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 225 fuel assemblies. The total Unit 2 spent fuel pool and cask pit storage capacity is limited to no more than 1716 fuel assemblies.

5.7 DELETED

LCO 3.7.15 ACTION A ← Add proposed LCO 3.7.15 ACTION A → A02

SR 3.7.15.1 ← Add proposed SR 3.7.15.1 → M01

A01

~~TABLE 5.6-1~~ Table 3.7.15-1 (page 1 of 1)

Table 3.7.15-1

Minimum Burnup Coefficients

Fuel Type	Cooling Time (Years)	Coefficients		
		A	B	C
1	0	-33.4237	25.6742	-1.6478
2	0	-25.3198	14.3200	-0.4042
3	0	-23.4150	16.2050	-0.5500
4	0	-33.2205	24.8136	-1.5199
	2.5	-31.4959	23.4776	-1.4358
	5	-30.4454	22.7456	-1.4147
	10	-28.4361	21.2259	-1.2946
	15	-27.2971	20.3746	-1.2333
	20	-26.1673	19.4753	-1.1403
5	0	-24.8402	23.5991	-1.2082
	2.5	-22.9981	21.6295	-1.0249
	5	-21.8161	20.5067	-0.9440
	10	-20.0864	19.0127	-0.8545
	15	-19.4795	18.3741	-0.8318
	20	-18.8225	17.7194	-0.7985
6	0	-32.4963	25.3143	-1.5534
	2.5	-30.6688	23.6229	-1.4025
	5	-29.2169	22.5424	-1.3274
	10	-27.2539	21.0241	-1.2054
	15	-25.7327	19.8655	-1.1091
	20	-25.2717	19.5222	-1.1163
7	0	-24.6989	24.1660	-1.2578
	2.5	-23.0399	22.3047	-1.0965
	5	-21.2473	20.6553	-0.9403
	10	-20.1775	19.5506	-0.9015
	15	-19.4037	18.6626	-0.8490
	20	-18.3326	17.7040	-0.7526
8	0	-43.4750	11.6250	0.0000

NOTES:

1. To qualify in a “fuel type”, the burnup of a fuel assembly must exceed the minimum burnup “BU” calculated by inserting the “coefficients” for the associated “fuel type” and “cooling time” into the following polynomial function:

   $BU = A + B*E + C*E^2$, where:

   BU = Minimum Burnup (GWD/MTU)

   E = Maximum Initial Planar Average Enrichment (weight percent U-235)

   A, B, C = Coefficients for each fuel type

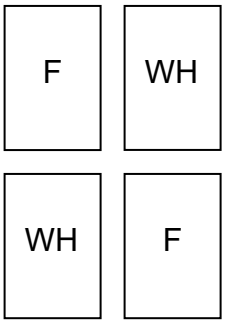
2. Interpolation between values of cooling time is not permitted.

Figure 3.7.15-1 (page 1 of 1)

Allowable Storage Patterns
(See Notes 1 and 2)

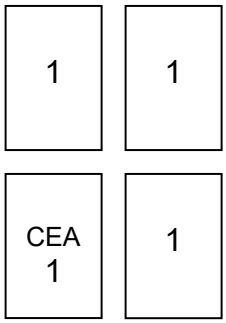
Pattern "A"
See ~~Definition 1~~

Note 3



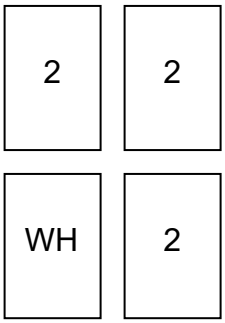
Pattern "B"
See ~~Definition 2~~

Note 4



Pattern "C"
See ~~Definition 3~~

Note 5



~~DEFINITIONS:~~

3 → ~~1.~~ Allowable pattern is fresh or burned fuel checkerboarded with completely water-filled cells. Diagram is for illustration only, where F represents Fuel and WH represents a completely water-filled cell.

4 → ~~2.~~ Allowable pattern is placement of fuel assemblies that meet the requirements of type 1 in each 2x2 array location with at least one full-length full-strength CEA placed in any cell. Minimum burnup for fuel assembly type 1 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

3.7.15-1

5 → ~~3.~~ Allowable pattern is placement of fuel assemblies that meet the requirements of type 2 in three of the 2x2 array locations in combination with one completely water-filled cell. Minimum burnup for fuel assembly type 2 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

3.7.15-1

NOTES:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Completely water-filled cells within any pattern are acceptable.

Figure 3.7.15-1 (page 1 of 1)

Figure 3.7.15-1 (page 1 of 1)

~~FIGURE 5.6-1~~

Allowable Region 1 Storage Patterns and Fuel Arrangements

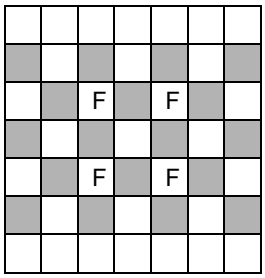
A01

Figure 3.7.15-2 (page 1 of 3)

ALLOWED SPECIAL ARRANGEMENTS (See Notes 1 and 2)
Fresh Fuel Assemblies in Region 2 Racks

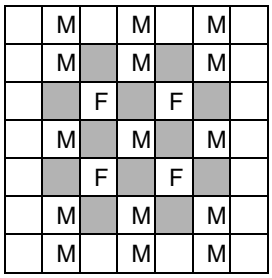
Note

See ~~Definition~~ SR1



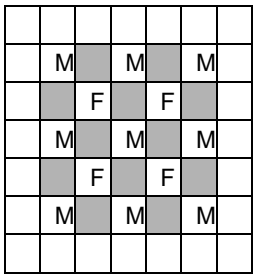
Note

See ~~Definition~~ SR2



Note

See ~~Definition~~ SR3



F = Fresh Fuel Assembly

M = Fuel Assembly with Metamic Insert

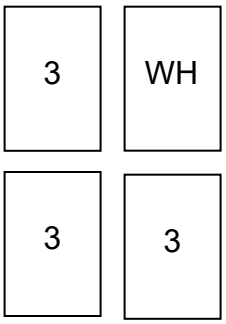
= Empty Cell

= Fuel Assembly per ~~Definition~~

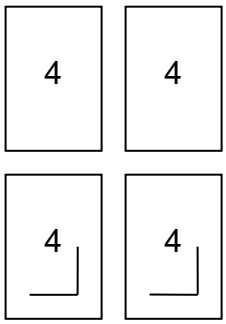
Notes

ALLOWABLE STORAGE PATTERNS (See Notes 1 and 2)

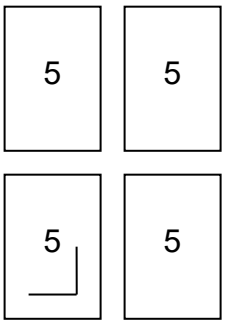
Pattern "D" Note 3
See ~~Definition~~ 1



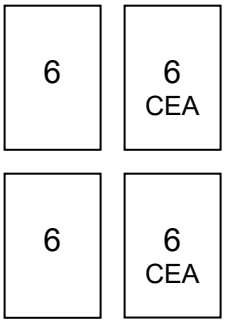
Pattern "E" Note 4
See ~~Definition~~ 2



Pattern "F" Note 5
See ~~Definition~~ 3



Pattern "G" Note 6
See ~~Definition~~ 4



Pattern "H" Note 7
See ~~Definition~~ 5

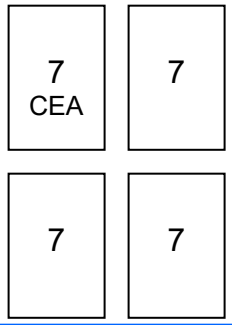


Figure 3.7.15-2 (page 1 of 3)

~~FIGURE 5.6-2 (Sheet 1 of 3)~~

Figure 3.7.15-2 (page 1 of 3)

Allowable Region 2 Storage Patterns and Fuel Arrangements

~~DEFINITIONS for Figure 5.6-2~~

Figure 3.7.15-2 (page 2 of 3)

Notes for Figure 3.7.15-2

3 → ~~1~~. Allowable pattern is fuel assemblies that meet the requirements of type 3 in three of the 2x2 array locations in combination with one completely water-filled cell. Minimum burnup for fuel assembly type 3 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

3.7.15

4 → ~~2~~. Allowable pattern is fuel assemblies that meet the requirements of type 4 in each of the 2x2 array locations with at least two Metamic inserts placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 4 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

3.7.15

5 → ~~3~~. Allowable pattern is fuel assemblies that meet the requirements of type 5 in each of the 2x2 array locations with at least one Metamic insert placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 5 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

3.7.15

6 → ~~4~~. Allowable pattern is fuel assemblies that meet the requirements of type 6 in each of the 2x2 array locations with at least two full-length, full strength 5 finger CEAs placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 6 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

3.7.15

7 → ~~5~~. Allowable pattern is fuel assemblies that meet the requirements of type 7 in each of the 2x2 array locations with at least one full-length, full strength 5 finger CEA placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 7 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

3.7.15

SR1. Allowable pattern is up to four fresh or burned fuel assemblies placed in a 3x3 array in combination with Pattern “D” placed outside the 3x3 array. Fresh or burned fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly that meets the requirements of type 3 shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 3 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

3.7.15

SR2. Allowable pattern is up to four fresh or burned fuel assemblies placed in a 3x3 array in combination with Pattern “E” placed outside the 3x3 array. Fresh or burned fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly that meets the requirements of type 4 with a Metamic insert shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 4 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

3.7.15

Figure 3.7.15-2 (page 2 of 3)

Figure 3.7.15-2 (page 2 of 3)

~~FIGURE 5.6-2 (Sheet 2 of 3)~~

Allowable Region 2 Storage Patterns and Fuel Arrangements

A01

Notes for Figure 3.7.15-2 (continued)

Figure 3.7.15-2 (page 3 of 3)

SR3. Allowable pattern is up to four fresh or burned fuel assemblies placed in a 3x3 array in combination with Pattern “F” placed outside the 3x3 array. Fresh or burned fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly that meets the requirements of type 5 with a Metamic insert shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 5 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

3.7.15

NOTES

1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Completely water-filled cells within any pattern are acceptable.

Figure 3.7.15-2 (page 3 of 3)

Figure 3.7.15-2 (page 3 of 3)

~~**FIGURE 5.6-2 (Sheet 3 of 3)**~~

Allowable Region 2 Storage Patterns and Fuel Arrangements

A01

Figure 3.7.15-3 (page 1 of 2)

Allowed Region 2 to Region 1 Fuel Alignments
(See Note 1)

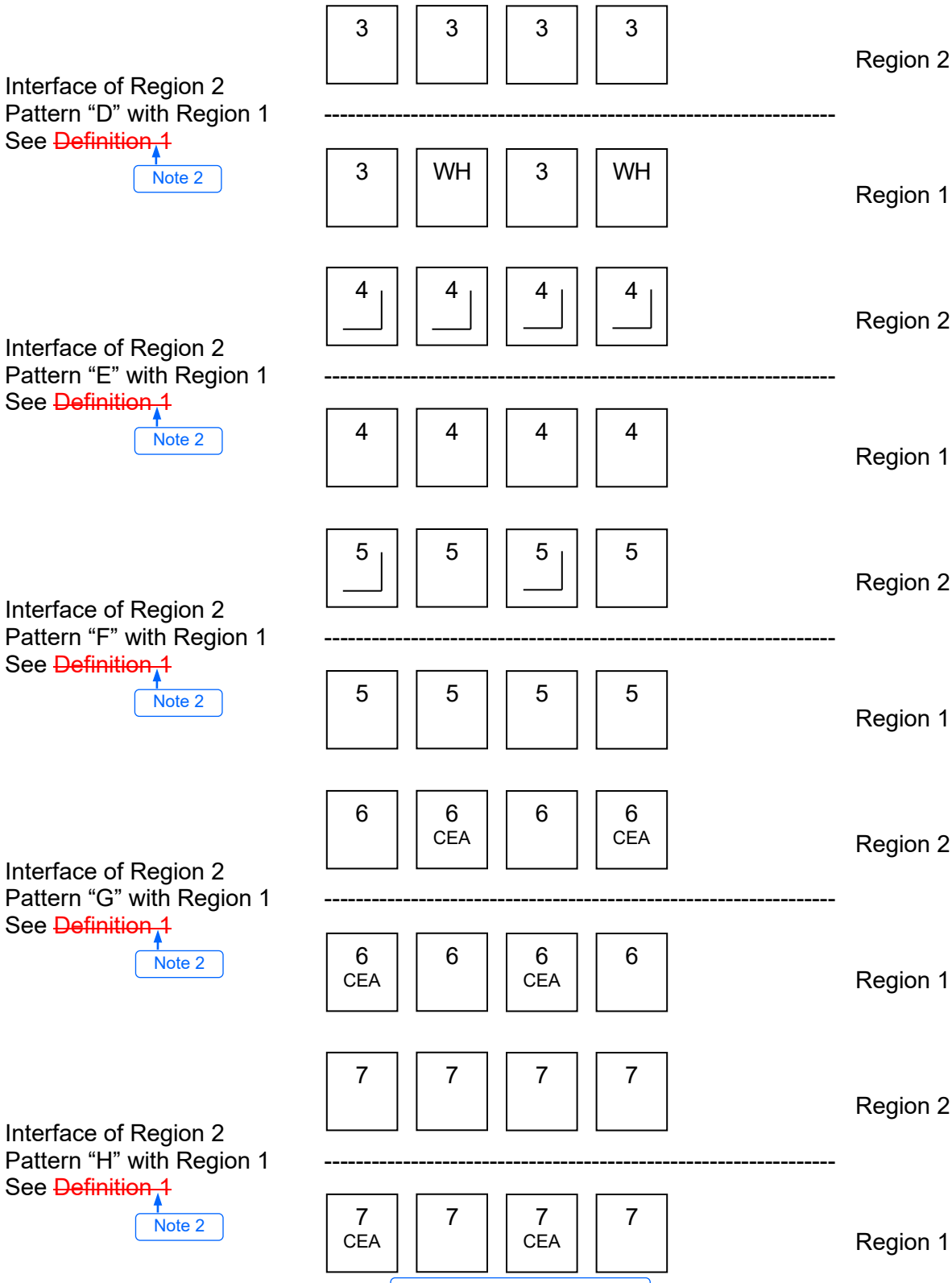


Figure 3.7.15-3 (page 1 of 2)

~~FIGURE 5.6-3 (Sheet 1 of 2)~~

Figure 3.7.15-3 (page 1 of 2)

Interface Requirements between Region 1 and Region 2

Figure 3.7.15-3 (page 2 of 2)

~~DEFINITION:~~

2 ~~1~~. Each 2x2 array that spans Region 1 and Region 2 shall match one of the Region 2 allowable storage patterns as defined by Specification ~~5.6.1.c.3~~. Any required Metamic inserts must be placed into the fuel assemblies in Region 2. Locations of completely water-filled cells or CEAs may be in either Region 1 or Region 2. For interface assemblies, the requirements of Specification~~s~~ ~~5.6.1.c.2~~ and Specification ~~5.6.1.c.3~~ shall be followed within Region 1 and Region 2, respectively. The Diagrams are for illustration only.

3.7.15.c

3.7.15.b

3.7.15.c

NOTES:

1. Completely water-filled cells within any pattern are acceptable.

Figure 3.7.15-3 (page 2 of 2)

Figure 3.7.15-3 (page 2 of 2)

~~FIGURE 5.6-3 (Sheet 2 of 2)~~

Interface Requirements between Region 1 and Region 2

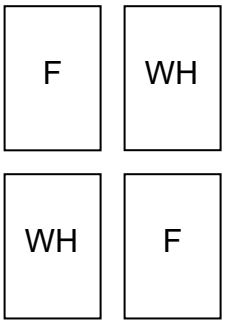
A01

Figure 3.7.15-4 (page 1 of 1)

Allowable Storage Patterns
(See Notes 1 and 2)

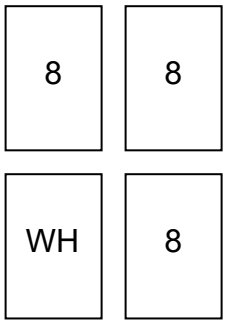
Pattern “A”
See ~~Definition 1~~

Note 3



Pattern “B”
See ~~Definition 2~~

Note 4



~~DEFINITIONS:~~

3 → ~~1.~~ Allowable pattern is fresh or burned fuel checkerboarded with completely water-filled cells. Diagram is for illustration only, where F represents Fuel and WH represents a completely water-filled cell.

4 → ~~2.~~ Allowable pattern is placement of fuel assemblies that meet the requirements of type 8 in three of the 2x2 array locations in combination with one completely water-filled cell in any location. Minimum burnup for fuel assembly type 8 is defined in Table ~~5.6~~-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

3.7.15

NOTES:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Completely water-filled cells within any pattern are acceptable.

Figure 3.7.15-4 (page 1 of 1)

Figure 3.7.15-4 (page 1 of 1)

~~FIGURE 5.6-4~~

Allowable Cask Pit Storage Rack Patterns

DISCUSSION OF CHANGES
ITS 3.7.15, SPENT FUEL POOL STORAGE

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 5.6.1.c provides the criteria for fuel storage in the spent fuel storage pool, based on enrichment and burnup, for Region 1, Region 2, or cask pit storage racks. Unit 1 ITS LCO 3.7.15 requires that the initial enrichment and burnup of each fuel assembly stored in Region 1 or Region 2 be in accordance with Table 3.7.15-1 and the configuration requirements specified in LCO 3.7.15.a, b, c, d, e and f. Unit 2 ITS LCO 3.7.15 requires that the initial enrichment and burnup of each fuel assembly stored in Region 1, Region 2, or cask pit storage rack be in accordance with Table 3.7.15-1 and the configuration requirements specified in LCO 3.7.15.a, b, c, d, e, f, and g. In addition, ITS 3.7.15 provides an explicit ACTION to initiate action to immediately move the noncomplying fuel assembly to an acceptable location if the requirements of the LCO are not met. This changes the CTS by moving the storage requirements for spent fuel stored in Region 1, Region 2, and cask pit storage racks to an explicit LCO and adds an explicit ACTION to be taken if the LCO is not met.

The purpose of CTS 5.6.1 is to provide the design criteria to preserve assumptions in the spent fuel storage pool criticality analysis. Spent fuel storage requirements are more appropriately provided in an LCO. Therefore, the spent fuel storage requirements specified in CTS 5.6.1.c, including associated table and figures are moved to a separate Technical Specifications; ITS 3.7.15, “Spent Fuel Storage.” Although the CTS does not provide an explicit Action associated with noncompliance with the storage requirements of CTS 5.6.1.c, this condition would result in the spent fuel storage pool being in an unanalyzed condition and immediate corrective action would be taken to restore compliance. This change is acceptable because the ITS preserves the assumptions of the spent fuel storage pool criticality analysis and provides an appropriate ACTION to restore compliance when fuel storage is not within the required configuration. This change is designated as an administrative change because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 The CTS does not provide a Surveillance Requirement for spent fuel storage. ITS SR 3.7.15.1 requires a verification by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.15-1 and configuration requirements of LCO 3.7.15.a, b, c, d, e, f and g (Unit 2 only). This changes the CTS by incorporating the requirements of ITS SR 3.7.15.1.

DISCUSSION OF CHANGES
ITS 3.7.15, SPENT FUEL POOL STORAGE

The safety related function of the spent fuel storage pool is to assure by the spacing and geometrically safe configurations of new and spent fuel assemblies with initial planar average U-235 enrichment up to 4.6 wt% ensure a subcritical array of keff ≤ 0.95 is maintained, assuming credit for a portion of the soluble boron present in the fuel pool water. This change is acceptable because the proposed Surveillance Requirement provides assurance that fuel assembly storage will be controlled in accordance with the assumptions of the spent fuel storage pool criticality analysis. This change is designated as more restrictive because it adds a new Surveillance Requirement to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7.~~18~~ Spent Fuel Pool Storage
15
3

5.6.1.c
DOC A02

LCO 3.7.~~18~~ The combination of initial enrichment and burnup of each fuel assembly stored in ~~[Region 2]~~ shall be ~~within the acceptable [burnup domain] of Figure 3.7.18-1 [or~~ in accordance with ~~Specification 4.3.1.1].~~
15
Region 1, Region 2, or cask pit
the following configuration requirements:
3
2

5.6.1.c.1, c.2, c.3, c.4, c.5, c.6
Insert 1
2

DOC A02

APPLICABILITY: Whenever any fuel assembly is stored in ~~[Region 2~~] of the fuel storage pool.
Region 1, Region 2, or cask pit storage rack
2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC A02 A. Requirements of the LCO not met.	A.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly ~~from [Region 2]~~. to an acceptable location	Immediately 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DOC M01 SR 3.7.~~18~~.1 15	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure 3.7.18-1 or Specification 4.3.1.1~~. Table 3.7.15-1 and the configuration requirements of LCO 3.7.15.a, b, c, d, e and f.	Prior to storing the fuel assembly in ~~[Region 2]~~ Region 1, Region 2, or cask pit storage rack 3 2

Insert 2 (Table 3.7.15-1)
Insert 3 (Figure 3.7.15-1)
Insert 4 (Figure 3.7.15-2)
Insert 5 (Figure 3.7.15-3)
A02

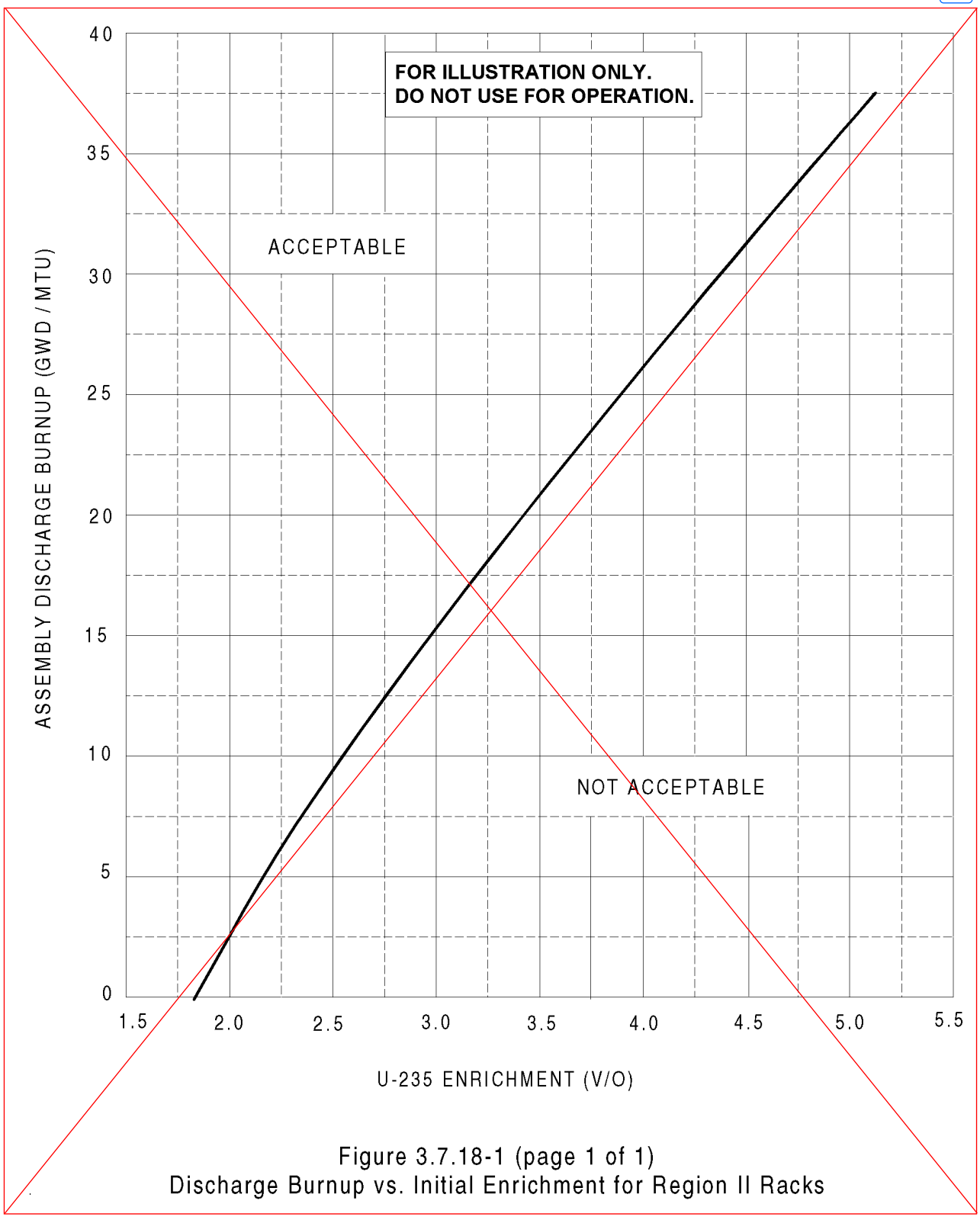


Figure 3.7.18-1 (page 1 of 1)
Discharge Burnup vs. Initial Enrichment for Region II Racks

1

CTS

Insert 1 (4)

5.6.1.c.1 a. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel storage rack shall be less than or equal to 4.6 weight percent.

5.6.1.c.2 b. Fuel placed in Region 1 of the spent fuel pool storage racks shall comply with the storage patterns and alignment restrictions of Figure 3.7.15-1 and the minimum burnup requirements of Table 3.7.15-1;

5.6.1.c.3 c. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage patterns or allowed special arrangements of Figure 3.7.15-2 and the minimum burnup requirements of Table 3.7.15-1;

----------NOTE----------
The allowed special arrangement for fresh fuel may be repeated provided the applicable interface requirements specified by the safety analysis are met.

5.6.1.c.4 d. Any fuel satisfying Specification 3.7.15.a, including fresh fuel, may be placed in the Region 1 cask pit storage rack.

----------NOTE----------
LCO 3.7.15.b, c, e, and f, do not apply to the Region 1 cask pit storage rack.

5.6.1.c.5 e. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required; and

5.6.1.c.6 f. Any 2x2 array of Region 2 storage cells that interface with Region 1 shall comply with the requirements of Figure 3.7.15-3.

----------NOTE----------
The allowed special arrangement in Region 2 as shown in Figure 3.7.15-2 shall not be placed adjacent to Region 1.

1

INSERT 2

Table 3.7.15-1 (page 1 of 1)
Minimum Burnup as a Function of Enrichment

Fuel Type	Cooling Time (Years)	Coefficients		
		A	B	C
1	0	-36.6860	22.4942	-1.4413
2	0	-36.1742	16.6000	-0.8958
3	0	-34.7091	23.1361	-1.6204
4	0	-24.5145	21.3404	-1.2444
	2.5	-26.8311	22.5246	-1.5029
	5	-24.7233	20.9763	-1.3246
	10	-23.6285	19.9541	-1.2505
	15	-23.5458	19.9336	-1.3180
	20	-22.4382	19.2460	-1.2629
5	0	-8.1856	14.5275	-0.0719
	2.5	-11.8506	16.1475	-0.3969
	5	-16.5196	18.5309	-0.7837
	10	-13.6831	16.3475	-0.5844
	15	-12.5819	15.6175	-0.5656
	20	-12.6469	15.4575	-0.5906

Notes:

1. To qualify in a ”fuel type,” the burnup of a fuel assembly must exceed the minimum burnup “BU” calculated by inserting the “coefficients” for the associated “fuel type” and “cooling time” into the polynomial function:

   $BU = A + B*E + C*E^2$, where:

   BU = Minimum Burnup (GWD/MTU)

   E = Initial Maximum Planar Average Enrichment (weight percent uranium-235)

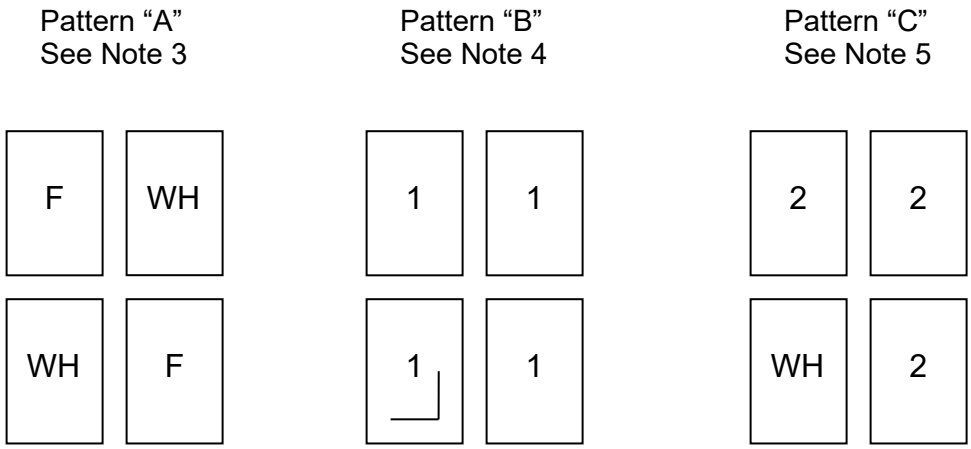
   A, B, C = Coefficients

2. Interpolation between values of cooling time is not permitted.

1

INSERT 3

Allowable Checkerboard Storage Patterns
(See Notes 1 and 2)



Notes:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Empty cells within any pattern are acceptable.

3. F represents Fresh Fuel. WH represents an empty cell. Allowable Pattern is Fresh Fuel checkerboarded with empty cells. Diagram is for illustration only.

4. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 1 is defined in Table 3.7.15-1. Allowable pattern is at least one insert [either Metamic or full-length full-strength CEA] in any one of the 2x2 array locations. Diagram is for illustration only.

5. Numbering denotes fuel assembly type. WH represents an empty cell. Minimum burnup for fuel assembly type 2 is defined in Table 3.7.15-1. Allowable pattern is at least one empty cell in any of the 2x2 array locations. Diagram is for illustration only.

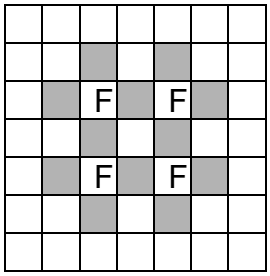
Figure 3.7.15-1 (page 1 of 1)
Allowable Region 1 Storage Patterns and Fuel Arrangements

1

INSERT 4

ALLOWED SPECIAL ARRANGEMENT

Fresh Fuel Assemblies in Pattern “C”, “D”, or “E” Racks

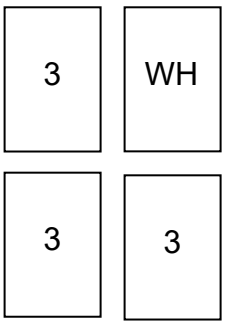


F = FRESH FUEL ASSEMBLY

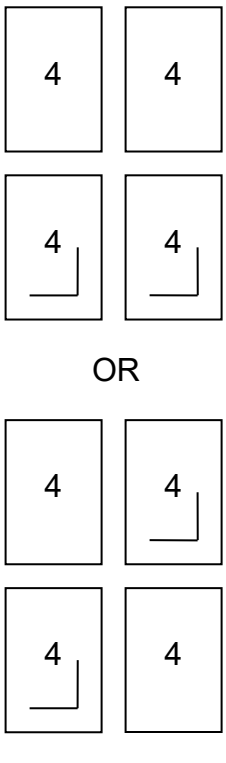
= EMPTY CELL

ALLOWABLE CHECKERBOARD STORAGE PATTERNS (See Notes 1 and 2)

Pattern “C”
See Note 3



Pattern “D”
See Note 4



Pattern “E”
See Note 5

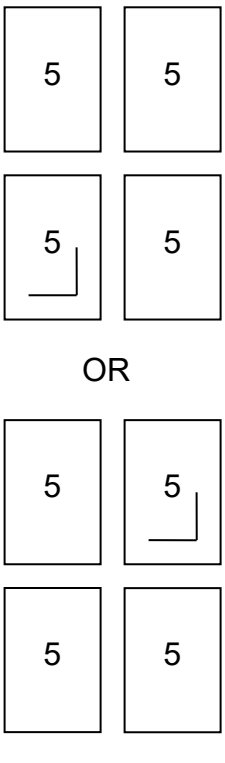


Figure 3.7.15-2 (page 1 of 2)
Allowable Region 2 Storage Patterns and Fuel Alignments

INSERT 4 (continued)

Notes to Figure 3.7.15-2

Notes:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Empty cells within any pattern are acceptable.

3. Numbering denotes fuel assembly type. WH represents an empty cell. Minimum burnup for fuel assembly type 3 is defined in Table 3.7.15-1. Allowable pattern is at least one empty cell in any of the 2x2 array locations. Diagram is for illustration only.

4. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 4 is defined in Table 3.7.15-1. Allowable pattern is at least two inserts, (either Metamic or full-length, full-strength CEA) in the 2x2 array. Diagrams are for illustration only.

5. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 5 is defined in Table 3.7.15-1. Allowable pattern is one insert, (either Metamic or full-length, full-strength CEA) in the 2x2 array. Diagrams are for illustration only.

Figure 3.7.15-2 (page 2 of 2)
Allowable Region 2 Storage Patterns and Fuel Arrangements

1

INSERT 5

Allowed Region 2 to Region 1 Fuel Alignments
(See Notes 1 and 2)

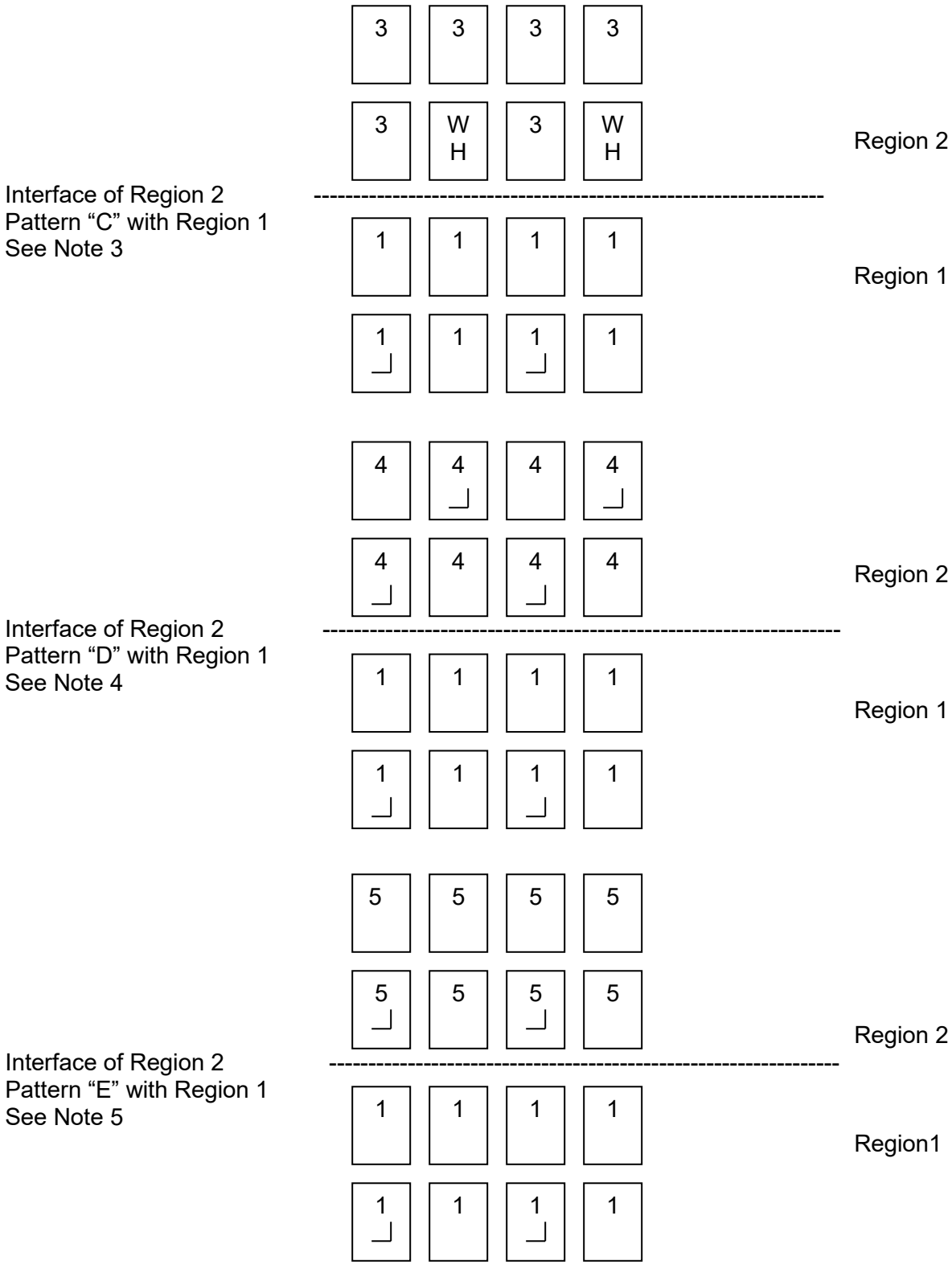


Figure 3.7.15-3 (page 1 of 2)
Region 2 Interface Requirements with Region 1

1

INSERT 5 (continued)

Notes to Figure 3.7.15-3

Notes:

1. Empty cells with any pattern are acceptable.

2. There are no interface requirements within Region 1. Any Pattern within Region 1 may be used for the interface. Pattern “B” was used only as an illustration.

3. WH represents an empty cell. For the interface of Pattern “C” with Region 1, the empty cell must be on the rack periphery facing Region 1 racks. Diagrams are for illustration only.

4. For the interface of pattern “D” with Region 1, at least one cell on the rack periphery facing Region 1 rack must contain an insert (either Metamic of full-length full-strength CEA) in the 2x2 array. If the insert is Metamic, the insert must be oriented so that the corner of the L-shape is located closest to the Region 1 rack. Diagram is for illustration only.

5. For the interface of Pattern “E” with Region 1, the insert must be on the rack periphery facing the Region 1 rack. The insert may be either a Metamic of full-length full strength CEA. If the insert is Metamic, the insert must be oriented so that the corner of the L-shape is located closest to the Region 1 rack. Diagram is for illustration only.

Figure 3.7.15-3 (page 2 of 2)
Region 2 Interface requirements with Region 1

3.7 PLANT SYSTEMS

3.7.~~18~~ Spent Fuel Pool Storage

15

3

5.6.1.c
DOC A02

LCO 3.7.~~18~~ 15

The combination of initial enrichment and burnup of each fuel assembly stored in ~~[Region 2]~~ shall be ~~within the acceptable [burnup domain] of Figure 3.7.18-1 [or~~ in accordance with ~~Specification 4.3.1.1].~~

Region 1, Region 2, or cask pit

the following configuration requirements:

3

2

5.6.1.c.1, c.2, c.3, c.4, c.5, c.6, c.7

Insert 1

2

DOC A02

APPLICABILITY: Whenever any fuel assembly is stored in ~~[Region 2~~] of the fuel storage pool.

Region 1, Region 2, or cask pit storage rack

2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC A02 A. Requirements of the LCO not met.	A.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly ~~from [Region 2]~~. to an acceptable location	Immediately

2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DOC M01 SR 3.7.~~18~~.1 15	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure 3.7.18-1 or Specification 4.3.1.1~~. Table 3.7.15-1 and the configuration requirements of LCO 3.7.15.a, b, c, d, e, f and g.	Prior to storing the fuel assembly in ~~[Region 2]~~ Region 1, Region 2, or cask pit storage rack

3

2

Insert 2 (Table 3.7.15-1)
Insert 3 (Figure 3.7.15-1)
Insert 4 (Figure 3.7.15-2)
Insert 5 (Figure 3.7.15-3)
Insert 6 (Figure 3.7.15-4)

2

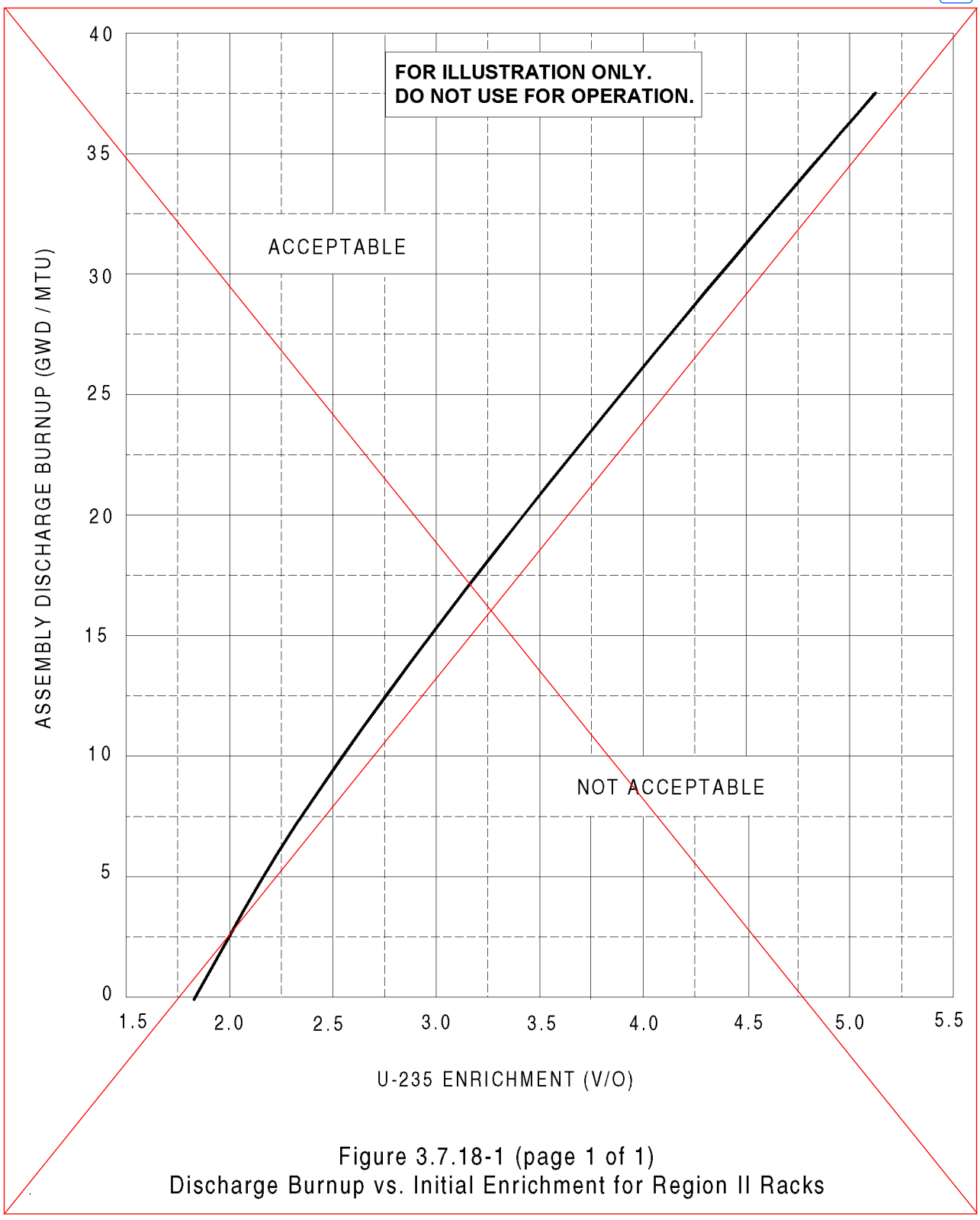


Figure 3.7.18-1 (page 1 of 1)
Discharge Burnup vs. Initial Enrichment for Region II Racks

1

CTS

Insert 1 (4)

5.6.1.c.1 a. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel storage rack shall be less than or equal to 4.6 weight percent.

5.6.1.c.2 b. Fuel placed in Region 1 of the spent fuel pool storage racks shall comply with the requirements of Figure 3.7.15-1 and the minimum burnup requirements of Table 3.7.15-1, except as specified in LCO 3.7.15.g;

5.6.1.c.3 c. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the requirements of Figure 3.7.15-2 and the minimum burnup requirements of Table 3.7.15-1, except as specified in LCO 3.7.15.g;

5.6.1.c.4 d. Any 2x2 array of fuel assemblies that span the interface between Region 1 and Region 2 of the spent fuel pool storage racks shall comply with the storage patterns of Figure 3.7.15-3 and the minimum burnup requirements of Table 3.7.15-1, except as specified in LCO 3.7.15.g;

----------NOTE----------
The allowed special arrangements in Region 2 as shown in Figure 3.7.15-2 shall not be placed adjacent to Region 1.

5.6.1.c.5 e. Fuel placed in the cask pit storage rack shall comply with the storage patterns of Figure 3.7.15-5 and the minimum burnup requirements of Table 3.7.15-1, except as specified in LCO 3.7.15.g;

5.6.1.c.6 f. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required;

5.6.1.c.7 g. Fresh or spent fuel in any allowed configuration may be replaced with non fuel hardware, and fresh fuel in any allowed configuration may be replaced with a fuel rod storage basket containing fuel rod(s).

----------NOTE----------
Storage of Metamic inserts or control rods, without any fissile material, is acceptable in locations designated as completely water-filled cells.

A01

INSERT 2

Table 3.7.15-1 (page 1 of 1)
Minimum Burnup Coefficients

Fuel Type	Cooling Time (Years)	Coefficients		
		A	B	C
1	0	-33.4237	25.6742	-1.6478
2	0	-25.3198	14.3200	-0.4042
3	0	-23.4150	16.2050	-0.5500
4	0	-33.2205	24.8136	-1.5199
	2.5	-31.4959	23.4776	-1.4358
	5	-30.4454	22.7456	-1.4147
	10	-28.4361	21.2259	-1.2946
	15	-27.2971	20.3746	-1.2333
	20	-26.1673	19.4753	-1.1403
5	0	-24.8402	23.5991	-1.2082
	2.5	-22.9981	21.6295	-1.0249
	5	-21.8161	20.5067	-0.9440
	10	-20.0864	19.0127	-0.8545
	15	-19.4795	18.3741	-0.8318
	20	-18.8225	17.7194	-0.7985
6	0	-32.4963	25.3143	-1.5534
	2.5	-30.6688	23.6229	-1.4025
	5	-29.2169	22.5424	-1.3274
	10	-27.2539	21.0241	-1.2054
	15	-25.7327	19.8655	-1.1091
	20	-25.2717	19.5222	-1.1163
7	0	-24.6989	24.1660	-1.2578
	2.5	-23.0399	22.3047	-1.0965
	5	-21.2473	20.6553	-0.9403
	10	-20.1775	19.5506	-0.9015
	15	-19.4037	18.6626	-0.8490
	20	-18.3326	17.7040	-0.7526
8	0	-43.4750	11.6250	0.0000

NOTES:

1. To qualify in a “fuel type”, the burnup of a fuel assembly must exceed the minimum burnup “BU” calculated by inserting the “coefficients” for the associated “fuel type” and “cooling time” into the following polynomial function:

   $BU = A + B*E + C*E^2$, where:

   BU = Minimum Burnup (GWD/MTU)

   E = Maximum Initial Planar Average Enrichment (weight percent U-235)

   A, B, C = Coefficients for each fuel type

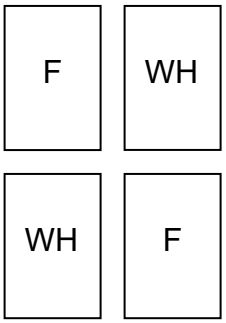
2. Interpolation between values of cooling time is not permitted.

1

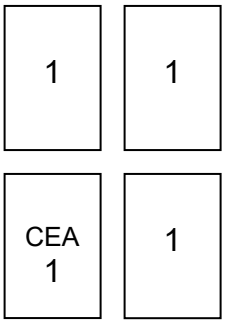
INSERT 3

Allowable Storage Patterns
(See Notes 1 and 2)

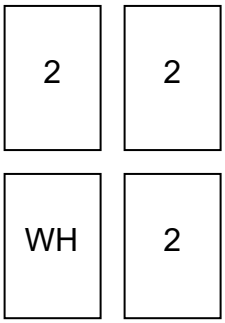
Pattern “A”
See Note 3



Pattern “B”
See Note 4



Pattern “C”
See Note 5



Notes:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Completely water-filled cells within any pattern are acceptable.

3. Allowable pattern is fresh or burned fuel checkerboarded with completely water-filled cells. Diagram is for illustration only, where F represents Fuel and WH represents a completely water-filled cell.

4. Allowable pattern is placement of fuel assemblies that meet the requirements of type 1 in each 2x2 array location with at least one full-length full-strength CEA placed in any cell. Minimum burnup for fuel assembly type 1 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

5. Allowable pattern is placement of fuel assemblies that meet the requirements of type 2 in three of the 2x2 array locations in combination with one completely water-filled cell. Minimum burnup for fuel assembly type 2 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

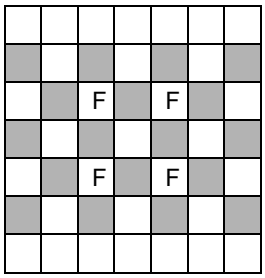
Figure 3.7.15-1 (page 1 of 1)
Allowable Region 1 Storage Patterns and Fuel Arrangements

1

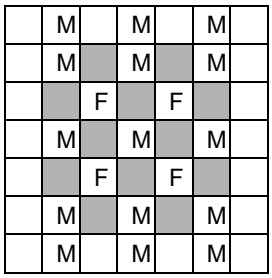
INSERT 4

ALLOWED SPECIAL ARRANGEMENTS (See Notes 1 and 2)
Fresh Fuel Assemblies in Region 2 Racks

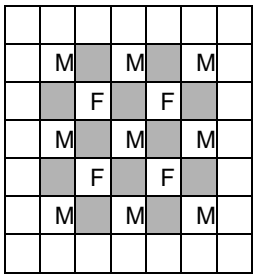
See Note SR1



See Note SR2



See Note SR3



F = Fresh Fuel Assembly

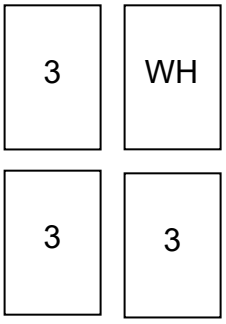
M = Fuel Assembly with Metamic Insert

= Empty Cell

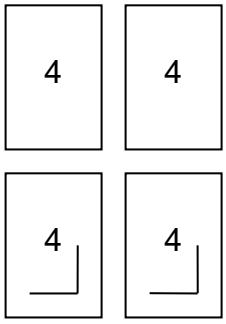
= Fuel Assembly per Notes

ALLOWABLE STORAGE PATTERNS (See Notes 1 and 2)

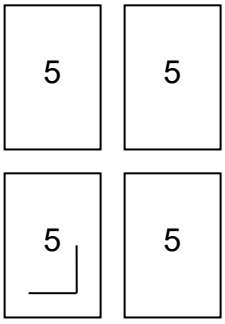
Pattern “D”
See Note 3



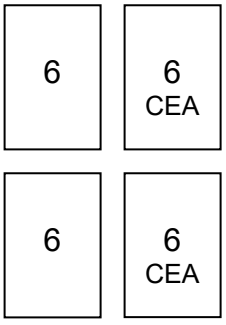
Pattern “E”
See Note 4



Pattern “F”
See Note 5



Pattern “G”
See Note 6



Pattern “H”
See note 7

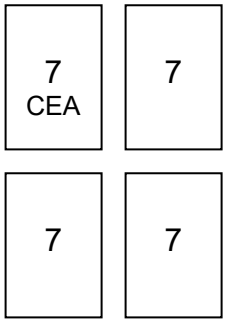


Figure 3.7.15-2 (page 1 of 3)
Allowable Region 2 Storage Patterns and Fuel Arrangements

1

INSERT 4 (continued)

Notes for Figure 3.7.15-2 (page 1 of 3)

Notes:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Completely water-filled cells within any pattern are acceptable.

3. Allowable pattern is fuel assemblies that meet the requirements of type 3 in three of the 2x2 array locations in combination with one completely water-filled cell. Minimum burnup for fuel assembly type 3 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

4. Allowable pattern is fuel assemblies that meet the requirements of type 4 in each of the 2x2 array locations with at least two Metamic inserts placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 4 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

5. Allowable pattern is fuel assemblies that meet the requirements of type 5 in each of the 2x2 array locations with at least one Metamic insert placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 5 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

6. Allowable pattern is fuel assemblies that meet the requirements of type 6 in each of the 2x2 array locations with at least two full-length, full strength 5 finger CEAs placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 6 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

7. Allowable pattern is fuel assemblies that meet the requirements of type 7 in each of the 2x2 array locations with at least one full-length, full strength 5 finger CEA placed anywhere in the 2x2 array. Minimum burnup for fuel assembly type 7 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

SR1. Allowable pattern is up to four fresh or burned fuel assemblies placed in a 3x3 array in combination with Pattern “D” placed outside the 3x3 array. Fresh or burned fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly that meets the requirements of type 3 shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 3 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

Figure 3.7.15-2 (page 2 of 3)
Allowable Region 2 Storage Patterns and Fuel Arrangements

INSERT 4 (continued)

Notes for Figure 3.7.15-2 (continued)

SR2. Allowable pattern is up to four fresh or burned fuel assemblies placed in a 3x3 array in combination with Pattern “E” placed outside the 3x3 array. Fresh or burned fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly that meets the requirements of type 4 with a Metamic insert shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 4 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

SR3. Allowable pattern is up to four fresh or burned fuel assemblies placed in a 3x3 array in combination with Pattern “F” placed outside the 3x3 array. Fresh or burned fuel shall be placed in the corners of the 3x3 array with completely water-filled cells placed face-adjacent on all sides. A fuel assembly that meets the requirements of type 5 with a Metamic insert shall be placed in the center of the 3x3 array. Minimum burnup for fuel assembly type 5 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment and cooling time. Diagram is for illustration only.

Figure 3.7.15-2 (page 3 of 3)
Allowable Region 2 Storage Patterns and Fuel Arrangements

1

INSERT 5

Allowed Region 2 to Region 1 Fuel Alignments
(See Note 1)

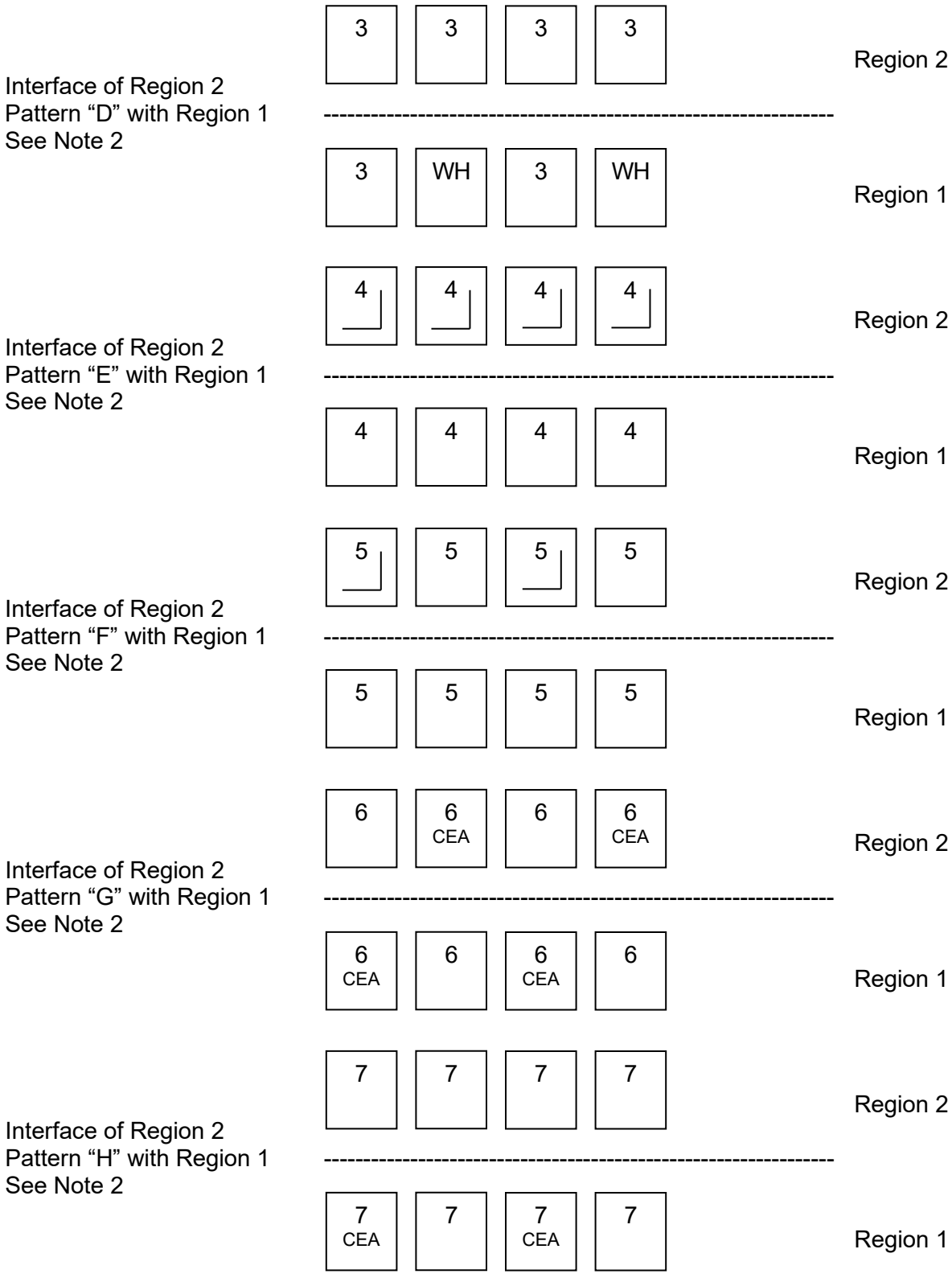


Figure 3.7.15-3 (page 1 of 2)
Interface Requirements between Region 1 and Region 2

INSERT 5 (continued)

Notes:

1. Completely water-filled cells within any pattern are acceptable.

2. Each 2x2 array that spans Region 1 and Region 2 shall match one of the Region 2 allowable storage patterns as defined by LCO 3.7.15.c. Any required Metamic inserts must be placed into the fuel assemblies in Region 2. Locations of completely water-filled cells or CEAs may be in either Region 1 or Region 2. For interface assemblies, the requirements of LCO 3.7.15.b and LCO 3.7.15.c shall be followed within Region 1 and Region 2, respectively. The Diagrams are for illustration only.

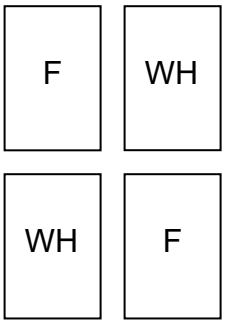
Figure 3.7.15-3 (page 2 of 2)
Interface Requirements between Region 1 and Region 2

1

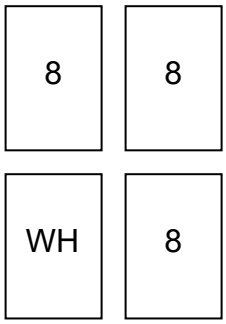
INSERT 6

Allowable Storage Patterns
(See Notes 1 and 2)

Pattern “A”
See Note 3



Pattern “B”
See Note 4



Notes:

1. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Completely water-filled cells within any pattern are acceptable.

3. Allowable pattern is fresh or burned fuel checkerboarded with completely water-filled cells. Diagram is for illustration only, where F represents Fuel and WH represents a completely water-filled cell.

4. Allowable pattern is placement of fuel assemblies that meet the requirements of type 8 in three of the 2x2 array locations in combination with one completely water-filled cell in any location. Minimum burnup for fuel assembly type 8 is defined in Table 3.7.15-1 as a function of maximum initial planar average enrichment. Diagram is for illustration only.

Figure 3.7.15-4 (page 1 of 1)
Allowable Cask Pit Storage Rack Patterns

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.15, SPENT FUEL POOL STORAGE

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the Specification Number and Title.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.~~18~~ 15 Spent Fuel Pool Storage

BASES

BACKGROUND

~~The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store [735] fuel assemblies, which includes storage for [15] failed fuel containers. The spent fuel storage cells are installed in parallel rows with center to center spacing of [12 31/32] inches in one direction, and [13 3/16] inches in the other orthogonal direction. This spacing and "flux trap" construction, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans, is sufficient to maintain a k_{eff} of ≤ 0.95 for spent fuel of original enrichment of up to [3.3]%. However, as higher initial enrichment fuel assemblies are stored in the spent fuel pool, they must be stored in a checkerboard pattern taking into account fuel burnup to maintain a k_{eff} of 0.95 or less.~~

INSERT 1

APPLICABLE SAFETY ANALYSES

~~The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.~~

INSERT 2

The spent fuel pool storage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to ~~[Figure 3.7.18-1],~~ in the ~~accompanying~~ LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 ~~assuming the pool to be flooded with unborated water~~. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool ~~according to [Figure 3.7.18-1], in the accompanying LCO~~. ~~Fuel assemblies not meeting the criteria of [Figure 3.7.18-1] shall be stored in accordance with Specification 4.3.1.1.~~

criteria for Region 1, Region 2, and cask pit specified

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in ~~[Region 2]~~ of the spent fuel pool.

Region 1, Region 2, or cask pit storage rack

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

Region 1, Region 2, or the cask pit storage rack

When the configuration of fuel assemblies stored in ~~[Region 2] the spent fuel pool~~ is not in accordance with ~~Figure [3.7.18-1]~~, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance ~~with Figure [3.7.18-1]~~.

criteria for Region 1, Region 2, and cask pit specified in the LCO

1

INSERT 1

The spent fuel storage pool is designed for the underwater storage of 1706 spent fuel assemblies plus 143 fuel assemblies in the cask pit storage rack, when installed. The total storage capability is 1849 assemblies with the cask pit rack installed.

The spent fuel storage pool high density spent fuel storage racks are divided into three separate and distinct regions. Region 1, with a maximum storage capacity of 342 fuel assemblies, is designed to accommodate new fuel with a maximum U-235 enrichment up to 4.95 weight percent (wt%) or spent fuel regardless of the discharge fuel burnup. Region 2, with a maximum storage capacity of 1364 fuel assemblies, is designed to accommodate high burnup fuel. The cask pit rack, with a maximum storage capacity of 143 fuel assemblies, is a Region 1 design capable of storing either new fuel or spent fuel regardless of the discharge fuel burnup. | R1

Criticality is precluded by the spacing and geometrically safe configurations of new and spent fuel assemblies with a maximum initial planar average U-235 enrichment up to 4.6 wt% to ensure a subcritical array of keff ≤ 0.95 is maintained, assuming 500 ppm of soluble boron is present in the fuel pool water.

1

INSERT 2

The misplacement of a fresh (unburned) fuel assembly with a minimum soluble poison of 500 ppm in the spent fuel storage pool could result in exceeding the regulatory limit of keff ≤ 0.95. This could possibly occur if a fresh fuel assembly of the highest permissible enrichment (4.6 wt%) were to be inadvertently misloaded into a Region 2 storage cell intended to be empty, or into a cell intended to hold a low reactivity fuel assembly. The reactivity consequences of these situations determined that the misloading of a fresh assembly into a cell intended to remain empty is the bounding condition. A boron concentration of 1500 ppm is required to ensure a keff ≤ 0.95 is maintained in the event of a misloaded fuel assembly.

BASES

ACTIONS (continued)

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~18~~.1

15

3

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure~~ [~~3.7.18-1~~] in the ~~accompanying~~ LCO. ~~For fuel assemblies in the unacceptable range of [Figure 3.7.18-1], performance of this SR will ensure compliance with Specification 4.3.1.1.~~

2

2

criteria for Region 1, Region 2, and cask pit specified

REFERENCES None.

B 3.7 PLANT SYSTEMS

B 3.7.~~18~~ Spent Fuel Pool Storage
15

BASES

BACKGROUND

~~The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store [735] fuel assemblies, which includes storage for [15] failed fuel containers. The spent fuel storage cells are installed in parallel rows with center to center spacing of [12 31/32] inches in one direction, and [13 3/16] inches in the other orthogonal direction. This spacing and "flux trap" construction, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans, is sufficient to maintain a k_{eff} of ≤ 0.95 for spent fuel of original enrichment of up to [3.3]%. However, as higher initial enrichment fuel assemblies are stored in the spent fuel pool, they must be stored in a checkerboard pattern taking into account fuel burnup to maintain a k_{eff} of 0.95 or less.~~

INSERT 1

APPLICABLE SAFETY ANALYSES

~~The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.~~

INSERT 2

The spent fuel pool storage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to ~~[Figure 3.7.18-1],~~ in the ~~accompanying~~ LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 ~~assuming the pool to be flooded with unborated water~~. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool ~~according to [Figure 3.7.18-1], in the accompanying LCO~~. ~~Fuel assemblies not meeting the criteria of [Figure 3.7.18-1] shall be stored in accordance with Specification 4.3.1.1.~~

criteria for Region 1, Region 2, and cask pit specified

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in ~~[Region 2]~~ of the spent fuel pool.

Region 1, Region 2, or cask pit storage rack

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

Region 1, Region 2, or the cask pit storage rack

When the configuration of fuel assemblies stored in ~~[Region 2] the spent fuel pool~~ is not in accordance with ~~Figure [3.7.18-1]~~, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance ~~with Figure [3.7.18-1]~~.

criteria for Region 1, Region 2, and cask pit specified in the LCO

1

INSERT 1

The spent fuel storage pool is designed for the underwater storage of 1491 spent fuel assemblies plus 225 fuel assemblies in the cask pit storage rack, when installed. The total storage capability is 1716 assemblies with the cask pit rack installed.

The spent fuel storage pool high density spent fuel storage racks are divided into three separate and distinct regions. Region 1, with a maximum storage capacity of 366 fuel assemblies, is designed to accommodate new fuel with a maximum U-235 enrichment up to 4.95 weight percent (wt%) or spent fuel regardless of the discharge fuel burnup. Region 2, with a maximum storage capacity of 1125 fuel assemblies, is designed to accommodate high burnup fuel. The cask pit rack, with a maximum storage capacity of 225 fuel assemblies, is a Region 2 design capable of storing spent fuel regardless of the discharge fuel burnup. | R1

Criticality is precluded by the spacing and geometrically safe configurations of new and spent fuel assemblies a maximum initial planar average U-235 enrichment up to 4.6 wt% to ensure a subcritical array of keff ≤ 0.95 is maintained, assuming 500 ppm of soluble boron is present in the fuel pool water.

1

INSERT 2

The misplacement of a fresh (unburned) fuel assembly with a minimum soluble poison of 500 ppm in the spent fuel storage pool could result in exceeding the regulatory limit of keff ≤ 0.95. This could possibly occur if a fresh fuel assembly of the highest permissible enrichment (4.6 wt%) were to be inadvertently misloaded into a Region 2 storage cell intended to be empty, or into a cell intended to hold a low reactivity fuel assembly. The reactivity consequences of these situations determined that the misloading of a fresh assembly into a cell intended to remain empty is the bounding condition. A boron concentration of 1500 ppm is required to ensure a keff ≤ 0.95 is maintained in the event of a misloaded fuel assembly.

BASES

ACTIONS (continued)

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~18~~.1

15

3

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with ~~Figure~~ [~~3.7.18-1~~] in the ~~accompanying~~ LCO. ~~For fuel assemblies in the unacceptable range of [Figure 3.7.18-1], performance of this SR will ensure compliance with Specification 4.3.1.1~~.

criteria for Region 1, Region 2, and cask pit specified

2

2

REFERENCES None.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.15, BASES, SPENT FUEL POOL STORAGE

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes are made to reflect the Specification Number and Title.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.15, SPENT FUEL POOL STORAGE

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 16

3.7.16, SECONDARY SPECIFIC ACTIVITY

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

PLANT SYSTEMS

ACTIVITY

SECONDARY SPECIFIC

LIMITING CONDITION FOR OPERATION

LCO 3.7.16 ~~3.7.1.4~~ The specific activity of the secondary coolant system shall be ≤ 0.10 μCi/gram DOSE EQUIVALENT I-131.

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTION:

ACTION A With the specific activity of the secondary coolant system > 0.10 μCi/gram DOSE EQUIVALENT I-131, be in ~~at least~~ ~~HOT STANDBY~~ within 6 hours and ~~in~~ ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

MODE 5

MODE 3

36

SURVEILLANCE REQUIREMENTS

is

SR 3.7.16.1 ~~4.7.1.4~~ The specific activity of the secondary coolant ~~system~~ ~~shall be determined to be~~ within ~~the~~ limit ~~by performing sampling and analysis as described in Table 4.7-2~~.

Verify

in accordance with the Surveillance Frequency Control Program

L01

A01

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY
1. Gross Activity Determination	SFCP
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.
	b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

L01

LA01

SR 3.7.16.1

SFCP

LA02

LA02

PLANT SYSTEMS

ACTIVITY

SECONDARY SPECIFIC

LIMITING CONDITION FOR OPERATION

LCO 3.7.16

~~3.7.1.4~~ The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcuries/gram DOSE EQUIVALENT I-131.

Applicability

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

ACTION A

With the specific activity of the secondary coolant system greater than 0.10 microcuries/gram DOSE EQUIVALENT I-131, be in ~~at least~~ ~~HOT STANDBY~~ within 6 hours and ~~in~~ ~~COLD SHUTDOWN~~ within ~~the following~~ ~~30~~ hours.

MODE 3

MODE 5

36

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

~~4.7.1.4~~ The specific activity of the secondary coolant ~~system shall be determined to be~~ within ~~the~~ limit ~~by performing sampling and analysis as described in Table 4.7-1~~.

is

Verify

in accordance with the Surveillance Frequency Control Program

L01

A01

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS

	TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1.	Gross Activity Determination	SFCP
SR 3.7.16.1 2.	Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.
		b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

L01

LA01

SFCP

LA02

LA02

DISCUSSION OF CHANGES
ITS 3.7.16, SECONDARY SPECIFIC ACTIVITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS Table 4.7-2 and Unit 2 CTS Table 4.7-1 Item 2 require an isotopic analysis to determine whether DOSE EQUIVALENT I-131 concentration is within limit. ITS SR 3.7.16.1 requires the verification that specific activity of the secondary coolant is within limit. This changes the CTS by moving the detail that an isotopic analysis must be performed to satisfy the requirements of the Surveillance to the Bases.

The removal of this detail for performing a Surveillance Requirement from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS SR 3.7.16.1 still retains the requirement to verify secondary coolant DOSE EQUIVALENT I-131 is within limit. Also, this change is acceptable because this type of procedural detail will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 (*Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program*) Unit 1 CTS Table 4.7-2 and Unit 2 CTS Table 4.7-1 Item 2.a require the DOSE EQUIVALENT I-131 sampling frequency to be once per 31 days whenever the gross activity determination indicates iodine concentration greater than 10% of the allowable limit. Unit 1 CTS Table 4.7-2 and Unit 2 CTS Table

4.7-1 Item 2.b allows the sampling frequency for the DOSE EQUIVALENT I-131 to be extended to once per 6 months whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limits. ITS SR 3.7.16.1 requires the DOSE EQUIVALENT I-131 to be verified at a periodic Frequency in accordance with the Surveillance Frequency Control Program (SFCP). This changes the CTS by relocating a Surveillance Frequency to the SFCP.

The purpose of this Surveillance is to assure that the necessary parameter is maintained within limits. The Frequency is proposed to be specified in accordance with the Surveillance Frequency Control Program and is acceptable because the periodic Frequency is placed under licensee control pursuant to the methodology described in NEI 04-10, which is specified in ITS Section 5.5. The SFCP provides the necessary administrative controls to require that surveillances related to testing, calibration and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Florida Power & Light Company has determined that the control of the periodic Frequency in accordance with the SFCP is consistent with the intent of TSTF-425, Revision 3, and with the NRC's model SE dated July 6, 2009 (74 FR 32001). The current sampling Frequencies specified in Items 2a and 2b of Unit 1 CTS Table 4.7-2 and Unit 2 CTS Table 4.7-1 represent periodic frequencies of 31 days versus 61 days based on iodine concentration determined during verification of gross activity. However, the requirement to verify gross activity is being deleted from the CTS (Refer to Discussion of Change (DOC) L01). Therefore, the periodic Frequency relocated to the SFCP for ITS SR 3.7.16.1 is 31 days. The ITS SR 3.7.16.1 frequency in accordance with the SFCP is a periodic frequency and does not meet the scope exclusion criteria identified in Section 1.0, "Introduction," of the TSTF-425 model SE.

PSL Unit 1 and Unit 2 adopted a SFCP in Amendment Nos. 223 (Unit 1) and 173 (Unit 2) (ADAMS Accession No. ML15127A066) as contained in CTS 6.8.4.o (Unit 1) and 6.8.4.q (Unit 2) (ITS 5.5.16). This change is designated as a less restrictive removal of detail change because the Surveillance Frequency is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 5 – Deletion of Surveillance Requirement)* Unit 1 CTS Table 4.7-2 and Unit 2 CTS Table 4.7-1 Item 1 require that the gross activity determination in accordance with the Surveillance Frequency Control Program. ITS 3.7.16 does not require any sampling to be performed to determine the gross activity of the secondary coolant. This changes the CTS by deleting the requirement for gross activity determination.

The purpose of Unit 1 CTS Table 4.7-2 and Unit 2 CTS Table 4.7-1 Item 1 is to determine the gross activity in order to determine the sampling Frequency for secondary coolant DOSE EQUIVALENT I-131. Based on the gross activity, the sample Frequency for determining DOSE EQUIVALENT I-131 can be extended to once per 6 months from once per 31 days. This change is acceptable

because the deleted Surveillance Requirement is not necessary to verify that the values used to meet the LCO are consistent with the safety analysis. Thus, appropriate values continue to be tested in a manner and at a Frequency necessary to give confidence that the assumptions in the safety analyses are protected. ITS SR 3.7.16.1 continues to require that the DOSE EQUIVALENT I-131 be determined within an acceptable periodic Frequency in accordance with the Surveillance Frequency Control Program. Refer to DOC LA02 for relocation of the DOSE EQUIVALENT I-131 Frequency. The secondary coolant DOSE EQUIVALENT I-131 is used in the accident analyses. The gross activity of the secondary coolant is not used in any accident analysis and is not relevant to the LCO or associated ACTIONS. This change is designated as less restrictive because a Surveillance that is required in the CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7. ~~19~~ Secondary Specific Activity

16

3.7.1.4 LCO 3.7. ~~19~~ The specific activity of the secondary coolant shall be ≤ ~~[~~0.10~~]~~ µCi/gm DOSE EQUIVALENT I-131.

16

2 3

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.1.4 Action A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.7.1.4 Table 4.7-2 DOC L01 DOC LA02 SR 3.7.~~19~~.1 Verify the specific activity of the secondary coolant is within limit. 16	~~[[31] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2 3

2

3.7 PLANT SYSTEMS

3.7. ~~19~~ Secondary Specific Activity

16

3.7.1.4 LCO 3.7. ~~19~~ The specific activity of the secondary coolant shall be ≤ ~~[~~0.10~~]~~ µCi/gm DOSE EQUIVALENT I-131.

16

2 3

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.7.1.4 Action A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
4.7.1.4 Table 4.7-1 DOC L01 DOC LA02 SR 3.7.~~19~~.1 Verify the specific activity of the secondary coolant is within limit. 16	~~[[31] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2 3

2

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.16, SECONDARY SPECIFIC ACTIVITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes have been made to reflect the change to the Specification number.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.~~19~~ 16 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives, and thus is indication of current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 µCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and reactor coolant LEAKAGE. Most of the isotopes have short half-lives (i.e., < 20 hours).

With the specified activity level, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about ~~[.13]~~ 0.27 rem should the main steam safety valves (MSSVs) open for the 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR ~~100~~ 50.67 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter ~~[15]~~ U 15.4.6 (Ref. 2), assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of ~~[~~0.10~~]~~ µCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates.

BASES

APPLICABLE SAFETY ANALYSES (continued)

With the loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through MSSVs and atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generator. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Shutdown Cooling System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through MSSVs and ADVs during the event.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ ~~[~~0.10~~]~~ µCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1). 2

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS, and contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~19~~ 16.1 3

This SR ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. ~~[The [31] day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

REFERENCES

1. 10 CFR ~~100.11~~ 50.67. 2
2. FSAR, Chapter ~~[15]~~ 15.4.6. 2

U 1

B 3.7 PLANT SYSTEMS

B 3.7.~~19~~ 16 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives, and thus is indication of current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 µCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and reactor coolant LEAKAGE. Most of the isotopes have short half-lives (i.e., < 20 hours).

With the specified activity level, the resultant 2 hour thyroid dose 0.27 to a person at the exclusion area boundary (EAB) would be about ~~[.13]~~ rem should the main steam safety valves (MSSVs) open for the 2 hours following a trip from full power.

Operating a unit at the allowable limits could 50.67 result in a 2 hour EAB exposure of a small fraction of the 10 CFR ~~100~~ (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The U accident analysis 15.1.6 of the main steam line break (MSLB), as discussed in the FSAR, Chapter ~~[15]~~ (Ref. 2), assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of ~~[~~0.10~~]~~ µCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates.

BASES

APPLICABLE SAFETY ANALYSES (continued)

With the loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through MSSVs and atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generator. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Shutdown Cooling System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through MSSVs and ADVs during the event.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ ~~[~~0.10~~]~~ µCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1). 2

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS, and contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~19~~ 16.1 3

This SR ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. ~~[The [31] day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

REFERENCES

1. 10 CFR ~~100.11~~ 50.67. 2
2. FSAR, Chapter ~~[15]~~ 15.1.6. 2

U 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.16, BASES, SECONDARY SPECIFIC ACTIVITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes have been made to reflect the change to the Specification number.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.16, SECONDARY SPECIFIC ACTIVITY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 17

3.7.17, Fuel Pool Area Ventilation System
Unit 1 Only

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

~~REFUELING OPERATIONS~~ — PLANT SYSTEMS

AREA

FUEL POOL VENTILATION SYSTEM ~~– FUEL STORAGE~~

LIMITING CONDITION FOR OPERATION

3.7.17

3.9.12 ~~At least~~ one fuel pool ventilation ~~system~~ shall be OPERABLE.

area

train

A02

L01

APPLICABILITY: ~~Whenever~~ recently irradiated fuel ~~is~~ in the spent fuel pool.

During movement of

storage

ACTION:

A02

area

trains

ACTION A

a. With no fuel pool ventilation ~~system~~ OPERABLE, suspend all operations involving movement of recently irradiated fuel within the spent fuel pool ~~or crane operation with loads over the recently irradiated spent fuel~~ ~~until at least one fuel pool ventilation system is restored to OPERABLE status.~~

storage

L02

L01

ACTIONS Note

b. The provisions of Specification 3.0.3 are not applicable.

A02

SURVEILLANCE REQUIREMENTS

train

area

4.9.12 The above required fuel pool ventilation ~~system~~ shall be demonstrated OPERABLE:

Operate required fuel pool are ventilation train

SR 3.7.17.1

a. In accordance with the Surveillance Frequency Control Program ~~by initiating flow through the HEPA filter and charcoal adsorber train~~ and verify~~ing~~ that the train operates for at least 15 minutes.

continuous

See ITS 5.5

b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

Insert proposed SR 3.7.17.2

A03

SURVEILLANCE REQUIREMENTS (Continued)

c. In accordance with the Surveillance Frequency Control Program by:

See ITS 5.5

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4.15 inches Water Gauge while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

SR 3.7.17.3

3. Verifying that the ~~ventilation system~~ maintains ~~the spent fuel storage pool area at~~ a negative pressure of ≥ ~~1/8~~ 0.125 inches Water Gauge relative to the outside atmosphere during ~~system~~ operation.

required fuel pool area ventilation train

train

A02

d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

See ITS 5.5

e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

~~**INTENTIONALLY DELETED**~~

~~DELETED~~

DISCUSSION OF CHANGES
ITS 3.7.17, FUEL POOL AREA VENTILATION SYSTEM
UNIT 1 ONLY

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.9.12 provides requirements for the fuel pool ventilation system. ITS LCO 3.7.17 provides similar requirements and specifically requires one fuel pool area ventilation train to be OPERABLE. CTS 3.9.12 utilizes the terms system and train interchangeably. The Fuel Pool Area Ventilation System is a subsystem of the Fuel Handling Building Ventilation System and consists of two exhaust fans and a common filter system. Therefore, for the purpose of this LCO, each exhaust fan and associated dampers and valves, including the common filter system, is defined as a train. The ACTIONS and Surveillance Requirements are modified to reflect this presentation preference and includes the word “required” where appropriate in accordance with the ISTS Writer’s Guide. This change is a presentation preference and designated as an administrative change. This change is acceptable because it does not result in a technical change to the CTS.

A03 CTS 4.9.12.b.1 through b.4 specify the fuel pool ventilation filter system Surveillances to be performed within the frequency requirements specified in CTS 4.9.12.b. CTS 4.9.12.1.c.1, c.2, d, and e also specify fuel pool ventilation filter system Surveillances to be performed within the frequency requirements specified in CTS 4.9.12.c, d, and e, respectively. ITS SR 3.7.17.2 requires performing required fuel pool area ventilation train filter testing in accordance with the Ventilation Filter Testing Program (VFTP). CTS 4.9.12 does not include a VFTP, but the requirements that make up the VFTP are retained in ITS Section 5.5. This changes the CTS by requiring testing in accordance with the VFTP and moving the specific filter requirements to ITS Section 5.5. This presentation change is acceptable because filter testing requirements are being retained in the VFTP as part of ITS Section 5.5, and ITS SR 3.7.17.2 references the VFTP for performing these tests. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS 3.7.17, FUEL POOL AREA VENTILATION SYSTEM
UNIT 1 ONLY

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 (*Category 2 – Relaxation of Applicability*) CTS 3.9.12 states that the requirements on Fuel Pool Ventilation System – Fuel Storage are applicable "Whenever recently irradiated fuel is in the spent fuel pool." ITS 3.7.17 is applicable "During movement of recently irradiated fuel assemblies in the spent fuel storage pool." This changes the CTS by restricting the Applicability of the Fuel Pool Area Ventilation System Specification to only when there is a potential for a fuel handling accident (FHA) involving recently irradiated fuel, that is, during the movement of recently irradiated fuel assemblies in the spent fuel storage pool. In addition, since the Applicability is now limited to when recently irradiated fuel is being moved, the portion of CTS Action a to restore at least one fuel pool ventilation system to OPERABLE status, after movement of fuel has been suspended, has also been deleted.

The purpose of CTS 3.9.12 is to ensure that the Fuel Pool Area Ventilation System can provide filtration to minimize the consequences of an FHA involving recently irradiated fuel (i.e., an irradiated fuel assembly that has occupied part of a critical reactor core within the previous 72 hours). This change is acceptable because the requirements continue to ensure that the structures, systems, and components are maintained in the specified conditions needed to minimize event consequences. The FHA analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. It is improbable to move irradiated fuel within 72 hours of a plant shutdown (i.e., keff < 0.99) because of the physical time required to perform a controlled plant shutdown, cooldown and depressurize the Reactor Coolant System (RCS), and disassemble the reactor vessel to access irradiated fuel to begin fuel movement. Additionally, filtration of the fuel handling building is not assumed for an FHA in the spent fuel storage pool. The ITS requires one train of the Fuel Pool Area Ventilation System to be OPERABLE to ensure that, in the unlikely event an FHA in the spent fuel storage pool occurs within 72 hours of a unit shutdown, fuel pool area ventilation and filtration are provided to minimize the consequences of this improbable event. This change is designated as less restrictive because the LCO requirements are applicable in fewer conditions than in the CTS.

L02 (*Category 4 – Relaxation of Required Action*) CTS 3.9.12 Action a states, in part, with no fuel pool ventilation system OPERABLE, suspend …crane operation with loads over the recently irradiated spent fuel. ITS 3.7.17 ACTIONS do not include

this action. This changes the CTS Actions by removing actions to suspend crane operation with loads over recently irradiated spent fuel.

The purpose of CTS 3.9.12 is to ensure that the Fuel Pool Area Ventilation System can provide filtration to minimize the consequences of an FHA involving recently irradiated fuel (i.e., an irradiated fuel assembly that has occupied part of a critical reactor core within the previous 72 hours). According to UFSAR Section 15.4.3 (Unit 1) 15.7.4.1.2 (Unit 2), an FHA is initiated by the dropping of an irradiated fuel assembly either in the containment or in the fuel building. The FHA analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. It is improbable to move irradiated fuel within 72 hours of a plant shutdown (i.e., keff < 0.99) because of the physical time required to perform a controlled plant shutdown, cooldown and depressurize the RCS, and disassemble the reactor vessel to access irradiated fuel to begin fuel movement and PSL procedurally restricts movement of irradiated fuel until 72 hours after shutdown. However, the ITS requires one train of the Fuel Pool Area Ventilation System to be OPERABLE to ensure that, in the unlikely event an FHA in the spent fuel storage pool occurs within 72 hours of a unit shutdown, fuel pool area ventilation and filtration are provided to minimize the consequences of this improbable event.

Crane travel related requirements were relocated from the CTS in PSL Unit 1 Amendment 190, ("St. Lucie Units 1 and 2 – Issuance of Amendments Regarding the Relocation of Spent Fuel Pool Crane Technical Specification Requirements (TAC NOS. MB5667 and MB5668)," dated April 28, 2004 (ADAMS Accession No. ML 040440111)) to the Updated Final Safety Analysis Report where the operational detail is controlled under 10 CFR 50.59 criteria. As cited in the NRC Safety Evaluation associated with Amendment 190, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," regulatory guidelines for control of heavy load lifts provide assurance of safe handling of heavy loads in areas where a load drop could impact stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. Section 5.1.1 of NUREG-0612 provides guidelines for reducing the likelihood of dropping heavy loads and provides criteria for establishing safe load paths; procedures for load-handling operations; training of crane operators; design, testing, inspection, and maintenance of cranes and lifting devices; and analyses of the impact of heavy load drops. The guidelines in Sections 5.1.2 through 5.1.6 address alternatives to either further reduce the probability of a load-handling accident or mitigate the consequences of heavy load drops. These alternatives include using a single-failure-proof crane to improve reliability through increased factors of safety and through redundancy or duality in certain active components. Criteria for design of single-failure-proof cranes are included in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants."

This change is acceptable because the possibility of damage to a recently irradiated fuel assembly as a consequence of mishandling components other than a recently irradiated fuel assembly is minimized by thorough training, detailed procedures and equipment design. The PSL crane design precludes the handling of heavy objects, such as shipping casks, over the spent fuel pool storage racks. Administrative controls prevent the movement of heavy loads

over the cask pit whenever the cask pit rack is installed in the cask area of the spent fuel storage pool. In addition, the cask handling crane design meets the regulatory guidance for single-failure-proof cranes in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants" and NUREG-0612. Administrative controls that control the movement of light loads or prevent movement of light loads over recently irradiated fuel assemblies are similar to those used for control of heavy loads, to the extent practicable, as advised in NUREG-0612. Consequently, the possibility of dropping a load other than a recently irradiated fuel assembly and damaging of recently irradiated fuel assemblies in the spent fuel storage pool is remote. Therefore, the CTS 3.9.12 action related to suspension of crane operation with loads over recently irradiated spent fuel is not necessary to be included in the technical specifications and is removed.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.7 PLANT SYSTEMS

3.9.12 3.7.1~~4~~ 7 Fuel ~~Building Air Cleanup~~ Pool Area Ventilation System ~~(FBACS)~~ 1

3.9.12 DOC A02 LCO 3.7.1~~4~~ 7 ~~Two FBACS~~ One fuel pool area ventilation train~~s~~ shall be OPERABLE. 2

~~----------NOTE----------~~
~~The fuel building boundary may be opened intermittently under administrative control.~~
~~----------~~ 3

3.9.12 Applicability DOC L01 APPLICABILITY: ~~[MODES 1, 2, 3, and 4,]~~
During movement of ~~[~~recently~~]~~ irradiated fuel assemblies in the spent fuel ~~building~~ storage pool. 4 1

ACTIONS

----------NOTE----------
3.9.12 Action b LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One FBACS train inoperable.~~	~~A.1 Restore FBACS train to OPERABLE status.~~	~~7 days~~
~~B. Two FBACS trains inoperable due to inoperable fuel building boundary in MODE 1, 2, 3, or 4.~~	~~B.1 Restore fuel building boundary to OPERABLE status.~~	~~24 hours~~

3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C. [Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.~~ ~~OR~~ ~~Two FBACS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.~~	~~C.1 Be in MODE 3.~~ ~~AND~~ ~~C.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours]~~
~~D. Required Action and Associated Completion Time [of Condition A] not met during movement of [recently] irradiated fuel assemblies in the fuel building.~~	~~D.1 Place OPERABLE FBACS train in operation.~~ ~~OR~~ ~~D.2 Suspend movement of [recently] irradiated fuel assemblies in the fuel building.~~	~~Immediately~~ ~~Immediately~~
~~E~~. ~~Two FBACS~~ train~~s~~ inoperable ~~during movement of [recently] irradiated fuel assemblies in the fuel building~~.	~~E~~.1 Suspend movement of ~~[~~recently~~]~~ irradiated fuel assemblies in the ~~fuel building~~.	Immediately

3
Required fuel pool area ventilation
A
A
3
4
1
3.9.12 Action a DOC L01 DOC L02
spent fuel storage pool

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.9.12.a	SR 3.7.14.1 Operate ~~each FBACS~~ train for ≥ 15 continuous minutes ~~[with heaters operating]~~.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
DOC A03	SR 3.7.14.2 Perform required ~~FBACS~~ filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~.	In accordance with the ~~[~~VFTP~~]~~
	~~SR 3.7.14.3 [Verify each FBACS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~
4.9.12.c.3 DOC A02	SR 3.7.14.4 Verify ~~one FBACS~~ train can maintain a negative pressure ≥ ~~[ ]~~ inches water gauge with respect to atmospheric pressure, during ~~the [post accident] mode of~~ operation ~~at a flow rate ≤ [3000] cfm~~.	~~[[18] months on a STAGGERED TEST BASIS~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.7.14.5~~	~~[Verify each FBACS filter bypass damper can be opened, except for dampers that are locked, sealed, or otherwise secured in the open position.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

4

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.17, FUEL POOL AREA VENTILATION SYSTEM
UNIT 1 ONLY

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The Fuel Building Ventilation System (ISTS 3.7.14) is renumbered as ITS 3.7.17 and renamed Fuel Pool Area Ventilation System. This results in renumbering the subsequent ISTS Specifications, as applicable, in the ITS. St. Lucie Plant (PSL) Unit 2 design does not utilize the Fuel Building Air Cleanup System (FBACS) for emergency filtration in the event of a fuel handling accident in the fuel handling building. Unit 2 utilizes the fuel pool area ventilation mode of the Shield Building Ventilation System for the emergency filtration function (ITS 3.6.9). Therefore, ISTS 3.7.14 is not included in the PSL Unit 2 ITS.

3. ISTS LCO 3.7.14 requires two FBACS trains to be OPERABLE and provides actions and Surveillances based on requiring both trains. ITS LCO 3.7.17 requires one fuel pool area ventilation train to be OPERABLE consistent with the CTS. The Fuel Pool Area Ventilation System is a subsystem of the Fuel Handling Building Ventilation System and consists of two exhaust fans and a common filter system. Since the Fuel Pool Area Ventilation System design utilizes a common filter and, as a result, does not support the single failure criterion, only one train is required to be OPERABLE. As a result, the ACTIONS applicable to one of two trains inoperable are not included in the ITS and the Surveillances are modified to require testing on the “required” fuel pool area ventilation train. Additionally, the ACTIONS applicable to MODES 1, 2, 3, and 4 are deleted since this Specification is not required in MODES 1, 2, 3, and 4 consistent with the CTS. Because the Specification is only required during movement of recently irradiated fuel assemblies in the spent fuel storage pool, ISTS 3.7.14, Condition E (ITS 3.7.17, Condition A) is modified to eliminate unnecessary reference to “during movement of [recently] irradiated fuel assemblies in the fuel building.” Because the Fuel Pool Area Ventilation System filters radioactivity in the spent fuel storage pool area and not the entire fuel handling building; and use of this Specification is unlikely (i.e., movement of recently irradiated fuel in the spent fuel storage pool is improbable), the Note to the LCO is not needed and not included in the ITS.

4. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

7 Pool Area Ventilation

B 3.7.1~~4~~ Fuel ~~Building Air Cleanup~~ System ~~(FBACS)~~

2

Fuel Pool Area Ventilation System

BASES

1

BACKGROUND

The ~~FBACS~~ filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident or loss of coolant accident. The ~~FBACS~~, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

exhaust fans and a common filter system. The common filter system

Fuel Pool Area Ventilation System

The ~~FBACS~~ consists of two ~~independent, redundant trains. Each train~~ consists of a ~~heater, a~~ prefilter ~~or demister~~, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and ~~a fan~~. Ductwork, valves or dampers, and instrumentation also form part of the system~~, as well as demisters, functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank. The downstream HEPA filter is not credited in the analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak.~~ The system initiates filtered ventilation of the fuel ~~handling building~~ following ~~receipt of a high radiation signal~~.

1

pool area

manual actuation from the control room

The ~~FBACS is a standby system~~, part of which may also be operated during normal unit operations. Upon ~~receipt of the actuating signal~~, ~~normal air discharges from the fuel handling building, the fuel handling building is isolated, and the stream of ventilation air discharges through the system filter trains~~. The prefilters ~~or demisters~~ remove any large particles in the air~~, and any entrained water droplets present,~~ to prevent excessive loading of the HEPA filters and charcoal adsorber~~s~~.

Fuel Pool Area Ventilation System is a subsystem of the Fuel Handling Building Ventilation System

Insert 1

U

6 and 12.2.2.5

Fuel Pool Area Ventilation System

The ~~FBACS~~ is discussed in the FSAR, Sections ~~[6.5.1], [~~9.4.~~5]~~, ~~and [15.7.4]~~ (Refs. 1, 2, and ~~3, respectively), because it may be used for normal, as well as post accident, atmospheric cleanup functions~~.

3

1

used

APPLICABLE SAFETY ANALYSES

The ~~FBACS~~ is ~~designed~~ to mitigate the consequences of a fuel handling accident ~~[~~involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~[X] days~~)~~]~~ in which ~~[~~all~~]~~ rods in the fuel assembly are assumed to be damaged. The analysis of the ~~fuel handling accident~~ is given in Reference 3. ~~The~~

(FHA)

72 hours

3

1

FHA

Insert 1 (1)

manual initiation, the fuel pool area air is exhausted through air inlets around the periphery of the spent fuel storage pool, through the common system filter, and discharged by two 100% capacity exhaust fans to the atmosphere via the fuel handling building vent stack.

APPLICABLE SAFETY ANALYSES (continued)

Insert 2 → ~~Design Basis Accident analysis of the fuel handling accident assumes that only one train of the FBACS is functional, due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).~~

Fuel Pool Area Ventilation System

The ~~FBACS~~ satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

minimize

1

LCO

One train of the Fuel Pool Area Ventilation System is

~~Two independent and redundant trains of the FBACS are~~ required to be OPERABLE to ~~ensure that at least one is available, assuming a single failure that disables the other train coincident with a loss of offsite power. Total system failure could result in the~~ atmospheric release from the fuel building ~~exceeding the 10 CFR 100 limits (Ref. 5)~~ in the ~~event of a fuel handling accident~~.

required fuel pool area ventilation train

spent fuel pool area

The ~~FBACS~~ is considered OPERABLE when the individual components necessary to control exposure in the ~~fuel handling building~~ are OPERABLE ~~in both trains~~. ~~An FBACS~~ train is considered OPERABLE when ~~its associated~~:

A fuel pool area ventilation

unlikely event of an FHA occurring within 72 hours of a unit shutdown

1

The exhaust

a. Fan is OPERABLE,

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions, and

c. ~~Heater, demister,~~ ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

~~The LCO is modified by a Note allowing the fuel building boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering and exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for fuel building isolation is indicated.~~

2

Insert 2 (1)

The FHA analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. Additionally, filtration of the fuel handling building is not assumed for an FHA in the spent fuel storage pool. The Technical Requirements Manual includes a decay time requirement that no fuel movement will commence until 72 hours after shutdown. This ensures that the FHA assumptions, including no filtration of the fuel handling building, are preserved. In the event an FHA in the spent fuel storage pool occurs within 72 hours of a unit shutdown, the Fuel Pool Area Ventilation System will provide filtration to minimize the consequences of this improbable event.

BASES

APPLICABILITY

~~In MODES 1, 2, 3, and 4, the FBACS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA (refer to LCO 3.7.13, "Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)") for units that use this system as part of their ECCS PREACS.~~ 2

spent fuel storage pool, one fuel pool area ventilation train

During movement of ~~[~~recently~]~~ irradiated fuel assemblies in the ~~fuel building, the FBACS~~ is required to be OPERABLE ~~to mitigate the consequences of a fuel handling accident [involving handling recently irradiated fuel~~. Due to radioactive decay, ~~FBACS~~ is only required to mitigate ~~fuel handling accidents involving~~ handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~[X] days~~)~~]~~. 3 2

FHAs that involve

72 hours

the Fuel Pool Area Ventilation System

1 3

Insert 3

~~In MODES 5 and 6, the FBACS is not required to be OPERABLE, since the ECCS is not required to be OPERABLE~~. 1

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

~~A.1~~

~~If one FBACS train is inoperable, action must be taken to restore OPERABLE status within 7 days. During this time period, the remaining OPERABLE train is adequate to perform the FBACS function. The 7 day Completion Time is reasonable, based on the risk from an event occurring requiring the inoperable FBACS train, and ability of the remaining FBACS train to provide the required protection.~~

~~B.1~~

~~REVIEWER'S NOTE~~
~~Adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of an intentional and unintentional entry into Condition B.~~

2

1

Insert 3

When movement of irradiated fuel assemblies within the spent fuel storage pool is not being conducted, the potential for an FHA within the fuel pool area does not exist. Additionally, due to radioactive decay, an FHA that involves handling non-recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core longer than 72 hours) will result in doses that are well within the guideline values specified in 10 CFR 50.67 even without fuel pool area ventilation filtration capability. Therefore, under these conditions the Fuel Pool Area Ventilation System is not required to be OPERABLE.

BASES

ACTIONS (continued)

If the fuel building boundary is inoperable in MODE 1, 2, 3, or 4, the FBACS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE fuel building boundary within 24 hours. During the period that the fuel building boundary is inoperable, appropriate compensatory measures [consistent with the intent, as applicable, of GDC 19, 60, 61, 63, 64 and 10 CFR Part 100] should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibility repair, and test most problems with the fuel building boundary.

[C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the Completion Time, or when both FBACS trains are inoperable for reasons other than an inoperable fuel building boundary (i.e., Condition B), the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.]

D.1 and D.2

When Required Action A.1 cannot be completed within the required Completion Time during movement of [recently] irradiated fuel assemblies in the fuel building, the OPERABLE FBACS train must be started immediately or fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected.

If the system is not placed in operation, this action requires suspension of [recently] irradiated fuel movement, which precludes a fuel handling accident. This does not preclude the movement of fuel to a safe position.

2

BASES

ACTIONS (continued)

A
one required fuel pool area ventilation train is
spent fuel storage pool

~~E~~.1

When ~~two trains of the FBACS are~~ inoperable during movement of ~~[~~recently~~]~~ irradiated fuel assemblies in the ~~fuel building~~, action must be taken to place the unit in a condition in which the LCO does not apply. This LCO involves immediately suspending movement of ~~[~~recently~~]~~ irradiated fuel assemblies in the ~~fuel building~~. This does not preclude the movement of fuel to a safe position.

2
3

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1
7
spent fuel storage pool
the required

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing ~~each~~ train ~~once every month~~ provides an adequate check on this system. Operation ~~[with the heaters on]~~ for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that ~~[heater failure,]~~ blockage, fan or motor failure, or excessive vibration can be detected for corrective action. ~~[The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.~~

1
periodically

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

3

SR 3.7.14.2
7
Fuel Pool Area Ventilation System

This SR verifies the performance of ~~FBACS~~ filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. The ~~[~~VFTP~~]~~ includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~.

1
2
3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[SR 3.7.14.3~~

~~This SR verifies that each FBACS train starts and operates on an actual or simulated actuation signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. [The [18] month Frequency is consistent with that specified in Reference 6.]~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]]~~

3

7.3

SR 3.7.1~~4~~4.4

2

pool area

system

Fuel Pool Area Ventilation System

This SR verifies the integrity of the fuel ~~building enclosure~~. The ability ~~of the fuel building~~ to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ~~FBACS~~. During ~~the post accident mode of~~ operation, the ~~FBACS~~ is designed to maintain a slight negative pressure in the fuel ~~building~~, with respect to adjacent areas, to prevent unfiltered LEAKAGE. ~~The FBACS is designed to maintain this negative pressure at a flow rate of ≤ [3000] cfm to the fuel building. [The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 7).~~

pool area

1

3

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~This test is conducted with the tests for filter penetration; thus, an [18] month Frequency, on a STAGGERED TEST BASIS is consistent with other filtration SRs.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

~~[SR 3.7.14.5~~

~~Operating the FBACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FBACS filter bypass damper is verified if it can be opened. The SR excludes automatic dampers that are locked, sealed, or otherwise secured in the open position. The SR does not apply to dampers that are locked, sealed, or otherwise secured in the open position since the affected dampers were verified to be in the open position prior to being locked, sealed, or otherwise secured. Placing an automatic damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the damper to be closed to support the accident analysis. Restoration of an automatic damper to the closed position requires verification that the SR has been met within its required Frequency. [The 18 month Frequency is consistent with that specified in Reference 6.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]]~~

3

BASES

REFERENCES

1. FSAR, Section [6.5.1].
2. FSAR, Section [9.4.5].
3. FSAR, Section [15.7.4].
4. Regulatory Guide 1.25.
5. 10 CFR 100.
6. Regulatory Guide 1.52, Rev. [2].
7. NUREG-0800, Section 6.5.1, July 1981.

JUSTIFICATION FOR DEVIATIONS
ITS 3.7.17 BASES, FUEL POOL AREA VENTILATION SYSTEM
UNIT 1 ONLY

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Changes have been made to be consistent with changes made to the Specifications.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.7.17, FUEL POOL AREA VENTILATION SYSTEM
UNIT 1 ONLY

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 18

Relocated/Deleted Current Technical Specifications

Unit 1 CTS 3/4.7.2, Steam Generator Pressure/Temperature Limitation R1
Unit 2 CTS 3/4.7.2, Steam Generator Pressure/Temperature Limitation

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: ALL MODES.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 200 psig within 30 minutes, and

b. Perform an analysis to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generators shall be determined to be < 200 psig at least once per hour when the temperature of either the primary or secondary coolant in the steam generators is < 70°F.

R1

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than 100°F when the pressure of the secondary coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure of the secondary side of the steam generators shall be determined to be less than 200 psig at least once per hour when the temperature of the secondary coolant is less than 100°F.

DISCUSSION OF CHANGES
UNIT 1 CTS 3/4.7.2, STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION
UNIT 2 CTS 3/4.7.2, STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

R1

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R01 Steam generator pressure/temperature limitation OPERABILITY requirements are intended to ensure that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations are based on a steam generator RTNDT and are sufficient to prevent brittle fracture.

The ITS does not include this Specification. This changes the CTS by relocating this Specification to the Technical Requirements Manual (TRM). This change is acceptable because the steam generator pressure/temperature limitation Specification does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. Steam generator P/T limits are ASME Code parameter limits and are not instrumentation used for, or capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary.

2. Steam generator P/T limits ae ASME Code parameter limits and do not represent a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Steam generator P/T limits are ASME Code parameter limits and not a structure, system, or component that is part of a primary success path in the mitigation of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Steam generator P/T limits and are not structures, systems, or components which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Since 10 CFR 50.36(c)(2)(ii) criteria have not been satisfied, the steam/generator pressure/temperature limitation may be relocated to a licensee controlled document outside the Technical Specifications. Requirements associated with the steam generator pressure/temperature limitation in support of safety related systems are addressed in ITS LCO 3.0.8. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation

R1

because the Specification does not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

R1

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
UNIT 1 CTS 3/4.7.2, STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION
UNIT 2 CTS 3/4.7.2, STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

There are no specific No Significant Hazards Considerations for this Specification.

Unit 2 CTS 3/4.7.6, Flood Protection

R1

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

R1

PLANT SYSTEMS

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for the facility site via stoplogs which shall be installed on the southside of the RAB and the southernmost door on east wall whenever a hurricane warning for the plant is posted.

APPLICABILITY: At all times.

ACTION:

With a Hurricane Watch issue for the facility site, ensure the stoplogs are removed from storage and are prepared for installation. The stoplogs shall be installed anytime a Hurricane Warning is posted.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Meteorological forecasts shall be obtained from the National Hurricane Center in Miami, Florida at least once per 6 hours during either a Hurricane Watch or a Hurricane Warning.

DISCUSSION OF CHANGES

UNIT 2 CTS 3/4.7.6, FLOOD PROTECTION

ADMINISTRATIVE CHANGES R1

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R01 External flooding during hurricane conditions is not designated as a PSL DBA or transient event. In addition, external flooding due to a hurricane is not postulated to occur during any DBA or transient, thus water level (as it pertains to wave run up effects during a hurricane) is not credited in any safety analysis. The Flood Protection Technical Specification requirements ensure that facility protective actions will be taken in the event of flood conditions whenever a hurricane warning is issued. The installation of the stoplogs ensures adequate protection for wave run-up effects where no permanent adjacent structures exist and provides protection to safety-related equipment.

The ITS does not include this Specification. This changes the CTS by relocating this Specification to the Technical Requirements Manual (TRM). This change is acceptable because the Flood Protection Specification does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. Flood protection requirements are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. Flood protection requirements are not process variables that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Flood protection requirements are not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The specific requirements specified in the Flood Protection Specification related to a hurricane are found to be non-significant risk contributors to core damage frequency and offsite releases and do not represent structures, systems, or components which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Since the screening criteria have not been satisfied, the Flood Protection Specification may be relocated to a licensee controlled document outside the Unit 2 Technical Specifications. A general requirement associated with hazard

R1

barriers (ITS LCO 3.0.9), which includes external flooding barriers, is added to ITS Section 3.0. A hazard barrier that cannot perform its related support function will be evaluated and managed under the Maintenance Rule plant configuration control requirement, 10 CFR 50.65(a)(4), and associated industry guidance, NUMARC 93-01, Revision 4A. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification does not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

R1

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

UNIT 2 CTS 3/4.7.6, FLOOD PROTECTION

There are no specific No Significant Hazards Considerations for this Specification.

Unit 1 CTS 3/4.7.10, Snubbers
Unit 2 CTS 3/4.7.9, Snubbers

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

PLANT SYSTEMS

3/4.7.10 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.10 All safety related snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more safety related snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.10 Each snubber shall be demonstrated OPERABLE by performance of the Snubber Testing Program.

DELETED

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

DELETED

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

DELETED

DELETED

.

LA01

PLANT SYSTEMS

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All safety-related snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more safety related snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9. on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the Snubber Testing Program.

a. Exemption From Visual Inspection or Functional Tests

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date.

LA01

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LA01

~~TABLE 3.7-3a~~

~~DELETED~~

LA01

TABLE 3.7-3b

DELETED

DISCUSSION OF CHANGES
UNIT 1 CTS 3/4.7.10, SNUBBERS
UNIT 2 CTS 3/4.7.9, SNUBBERS

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

LA01 (*Type 4 - Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPM, IST Program, or ISI Program*) Unit 1 CTS 3.7.10 and Unit 2 CTS 3.7.9 provide requirements for all safety-related snubbers. This specification, with the exception of the Action to restore an inoperable snubber within 72 hours, is not included in the ITS. This changes the CTS by moving the explicit snubber requirements from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details from the Technical Specification is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. The purpose of the snubber requirements is to ensure that the structural integrity of the Reactor Coolant System and other safety related systems is maintained during and following a seismic or other event initiating dynamic loads.

This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Specifically, ITS LCO 3.0.8 continues to require, in part, action to assess and manage risk during the period one or more required snubbers are unable to perform their associated support function(s). ITS LCO 3.0.8 also places limits on the delay period when one or more required snubbers are unable to perform their associated support function(s) based on the affect the degraded snubber has on the support system (e.g., affects one train or multiple trains). Refer to Section 3.0 Discussion of Changes related to the addition of ITS LCO 3.0.8. The requirement to perform snubber inspections is specified in 10 CFR 50.55a and the requirement to perform snubber inspections and testing is specified in ASME Section XI, as modified by approved relief requests. Therefore, both PSL commitments and NRC regulations or generic guidance contain the necessary programmatic requirements for the inspection and testing of safety related snubbers without repeating them in the ITS. ASME code requirements associated with snubber inspections and testing will continue to be controlled pursuant 10 CFR 50.55a Also, this change is acceptable because the removed information will be adequately controlled in the TRM. The TRM is incorporated by reference into the UFSAR, and any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a requirement is being removed from the Technical Specifications.

DISCUSSION OF CHANGES
UNIT 1 CTS 3/4.7.10, SNUBBERS
UNIT 2 CTS 3/4.7.9, SNUBBERS

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
UNIT 1 CTS 3/4.7.10, SNUBBERS
UNIT 2 CTS 3/4.7.9, SNUBBERS

There are no specific No Significant Hazards Considerations for this Specification.

Unit 1 CTS 3/4.7.9.1, Sealed Source Contamination
Unit 2 CTS 3/4.7.10, Sealed Source Contamination

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

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A01

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R01

PLANT SYSTEMS

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of ≥ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

1. Either decontaminated and repaired, or

2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1.1 Test Requirements – Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.9.1.2 Test Frequencies – Each category of sealed sources shall be tested at the frequencies described below.

a. Sources in use (excluding startup sources previously subjected to core flux) – In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive material:

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

b. Stored sources not in use – Each sealed source shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources – Each sealed startup source shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4.7.9.1.3 Reports – A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days if source leakage tests reveal the presence of ≥ 0.005 microcuries of removable contamination.

R01

PLANT SYSTEMS

3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.10 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, immediately withdraw the sealed source from use and either:

1. Decontaminate and repair the sealed source, or

2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.1 Test Requirements – Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall be a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.10.2 Test Frequencies – Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequencies described below.

a. Sources in use – In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive material:

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

R01

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use – Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certification indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors – Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source or detector.

4.7.10.3 Reports – A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

DISCUSSION OF CHANGES
UNIT 1 CTS 3/4.7.9.1, SEALED SOURCE CONTAMINATION
UNIT 2 CTS 3/4.7.10, SEALED SOURCE CONTAMINATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R01 The limitations on sealed source contamination are intended to ensure that the total body and individual organ irradiation doses do not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of < 0.005 microcuries of removable contamination on each sealed source. This requirement and the associated surveillance requirements bear no relation to the conditions or limitations that are necessary to ensure safe reactor operation.

The ITS does not include this Specification. This changes the CTS by relocating this Specification to the Technical Requirements Manual (TRM). This change is acceptable because the Seal Source Contamination Specification does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The sealed source contamination limitation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary.

2. The sealed source contamination limitation is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The sealed source contamination limitation is not part of a primary success path in the mitigation of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The sealed source contamination limitation is not discussed in the PSL PRA and is not a structure, system, or component which operating experience or

probabilistic risk assessment has shown to be significant to public health and safety.

Since 10 CFR 50.36(c)(2)(ii) criteria have not been satisfied, the Sealed Source Contamination Specification may be relocated to a licensee controlled document outside the Technical Specifications. Requirements associated with the sealed sources are governed by 10 CFR Part 70. Compliance with applicable portions of 10 CFR Part 70 is required by the operating licenses of PSL Units 1 and 2. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification does not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
UNIT 1 CTS 3/4.7.9.1, SEALED SOURCE CONTAMINATION
UNIT 2 CTS 3/4.7.10, SEALED SOURCE CONTAMINATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 19

ISTS Not Adopted

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7.10 Essential Chilled Water (ECW)

1

LCO 3.7.10 [Two] ECW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ECW train inoperable.	A.1 Restore ECW train to OPERABLE status.	7 days [OR In accordance with the Risk Informed Completion Time Program]
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.10.1	----------NOTE---------- Isolation of ECW flow to individual components does not render the ECW System inoperable. ---------- Verify each ECW manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	[31 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.7.10.2	Verify the proper actuation of each ECW System component on an actual or simulated actuation signal, except for valves that are locked, sealed, or otherwise secured in the actuated position.	[[18] months OR In accordance with the Surveillance Frequency Control Program]

1

JUSTIFICATION FOR DEVIATIONS
ISTS 3.7.10, ESSENTIAL CHILLED WATER (ECW)

1. This Essential Chilled Water (ECW) Specification is not needed because ECW is not included in the PSL Unit 1 and Unit 2 design and licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.10 Essential Chilled Water (ECW) System

1

BASES

BACKGROUND

The ECW System provides a heat sink for the removal of process and operating heat from selected safety related air handling systems during a Design Basis Accident (DBA) or transient.

The ECW System is a closed loop system consisting of two independent trains. Each 100% capacity train includes a heat exchanger, surge tank, pump, chemical addition tank, piping, valves, controls, and instrumentation. An independent 100% capacity chilled water refrigeration unit cools each train. The ECW System is actuated on a safety injection actuation signal (SIAS) and supplies chilled water to the heating, ventilation, and air conditioning (HVAC) units in Engineered Safety Feature (ESF) equipment areas (e.g., the main control room, electrical equipment room, and safety injection pump area).

The flow path for the ECW System includes the closed loop of piping to all serviced equipment, and branch lines up to the first normally closed isolation valve.

During normal operation, the normal HVAC System performs the cooling function of the ECW System. The normal HVAC System is a nonsafety grade system that automatically shuts down when the ECW System receives a start signal. Additional information about the design and operation of the system, along with a list of components served, can be found in the FSAR, Section [9.2.9] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the ECW System is to remove the post accident heat load from ESF spaces following a DBA coincident with a loss of offsite power. Each train provides chilled water to the HVAC units at the design temperature of 42°F and flow rate of 400 gpm.

The maximum heat load in the ESF pump room area occurs during the recirculation phase following a loss of coolant accident. During recirculation, hot fluid from the containment sump is supplied to the high pressure safety injection and containment spray pumps. This heat load to the area atmosphere must be removed by the ECW System to ensure that these pumps remain OPERABLE.

The ECW satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

[Two] ECW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst single failure. 1

An ECW train is considered OPERABLE when:

a. The associated pump and surge tank are OPERABLE and

b. The associated piping, valves, heat exchanger, refrigeration unit, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of the ECW from other components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the ECW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ECW System is required to be OPERABLE when a LOCA or other accident would require ESF operation.

In MODES 5 and 6, potential heat loads are smaller and the probability of accidents requiring the ECW System is low.

ACTIONS

A.1

If one ECW train is inoperable, action must be taken to restore OPERABLE status within 7 days [or in accordance with the Risk Informed Completion Time Program]. In this condition, one OPERABLE ECW train is adequate to perform the cooling function. The 7 day Completion Time is reasonable, based on the low probability of an event occurring during this time, the 100% capacity OPERABLE ECW train, and the redundant availability of the normal HVAC System.

B.1 and B.2

----------REVIEWER'S NOTE ----------
Adoption of a MODE 4 end state requires the licensee to make the following commitments:

1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].

2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion

BASES

ACTIONS (continued)

Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.

If the ECW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which overall plant risk is minimized. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Verifying the correct alignment for manual, power operated, and automatic valves in the ECW flow path provides assurance that the proper flow paths exist for ECW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

BASES

SURVEILLANCE REQUIREMENTS (continued)

1

This SR is modified by a Note indicating that the isolation of ECW flow to components or systems may render those components inoperable but does not affect the OPERABILITY of the ECW System.

[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------REVIEWER'S NOTE----------

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

----------]

SR 3.7.10.2

This SR verifies proper automatic operation of the ECW System components that the ECW pumps will start in the event of any accident or transient that generates an SIAS. This SR also ensures that each automatic valve in the flow paths actuates to its correct position on an actual or simulated SIAS. The SR excludes automatic valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to valves that are locked, sealed, or otherwise secured in the actuated position since the affected valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve to the non-actuated position requires verification that the SR has been met within its required Frequency. The ECW System cannot be fully actuated as part of the SIAS CHANNEL FUNCTIONAL TEST during normal operation. The actuation logic is tested as part of the SIAS functional test every 92 days, except for the subgroup relays that actuate the system that cannot be

BASES

SURVEILLANCE REQUIREMENTS (continued)

tested during normal unit operation. [The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The [18] month Frequency is based on operating experience and design reliability of the equipment.
OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------REVIEWER'S NOTE----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

REFERENCES

1. FSAR, Section [9.2.9].

2. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

JUSTIFICATION FOR DEVIATIONS
ISTS 3.7.10, BASES, ESSENTIAL CHILLED WATER (ECW)

1. ISTS 3.7.10, “Essential Chilled Water (ECW),” Specification and associated Bases are not included in the PSL ITS because ECW is not included in the PSL Unit 1 and Unit 2 design.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.7 PLANT SYSTEMS

3.7.15 Penetration Room Exhaust Air Cleanup System (PREACS)

1

LCO 3.7.15 Two PREACS trains shall be OPERABLE.

----------NOTE----------
The penetration room boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PREACS train inoperable.	A.1 Restore PREACS train to OPERABLE status.	7 days
B. Two PREACS trains inoperable due to inoperable penetration room boundary.	B.1 Restore penetration room boundary to OPERABLE status.	24 hours
C. ---------- NOTE ---------- Not applicable when second PREACS train intentionally made inoperable. ---------- Two PREACS trains inoperable for reasons other than Condition C.	C.1 Verify at least one train of containment spray is OPERABLE. AND C.2 Restore at least one ECCS PREACS train to OPERABLE status.	1 hour 24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Operate each PREACS train for ≥ 15 continuous minutes [with heater operating].	[31 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.7.15.2	Verify required PREACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.15.3	[Verify each PREACS train actuates on an actual or simulated actuation signal, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.	[[18] months OR In accordance with the Surveillance Frequency Control Program]]
SR 3.7.15.4	[Verify one PREACS train can maintain a negative pressure ≥ [ ] inches water gauge with respect to atmospheric pressure during the [post accident] mode of operation at a flow rate of ≤ [3000] cfm.	[[18] months on a STAGGERED TEST BASIS OR In accordance with the Surveillance Frequency Control Program]]
SR 3.7.15.5	[Verify each PREACS filter bypass damper can be opened, except for dampers that are locked, sealed, or otherwise secured in the open position.	[[18] months OR In accordance with the Surveillance Frequency Control Program]]

1

JUSTIFICATION FOR DEVIATIONS
ISTS 3.7.15, PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM (PREACS)

1. This Penetration Room Exhaust Air Cleanup System (PREACS) Specification is not needed because the PREACS is not included in the PSL Unit 1 and Unit 2 design and licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.7 PLANT SYSTEMS

B 3.7.15 Penetration Room Exhaust Air Cleanup System (PREACS)

1

BASES

BACKGROUND

The PREACS filters air from the penetration area between containment and the Auxiliary Building.

The PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters, which follows the adsorber section, collects carbon fines and provides backup in case of failure of the main HEPA filter bank. The downstream HEPA filter, although not credited in the accident analysis, collects charcoal fines and serves as a backup should the upstream HEPA filter develop a leak. The system initiates filtered ventilation following receipt of a safety injection actuation signal or containment isolation actuation signal.

The PREACS is a standby system, parts of which may also operate during normal unit operations. During emergency operations, the PREACS dampers are realigned, and fans are started to initiate filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the penetration room, the penetration room is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters or demisters remove any large particles in the air, as well as any entrained water droplets, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The PREACS is discussed in the FSAR, Sections [6.5.1], [9.4.5], and [15.6.5] (Refs. 1, 2, and 3, respectively), as it may be used for normal, as well as post accident, atmospheric cleanup functions. Heaters may be included for moisture removal on systems operating in high humidity conditions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level, consistent with iodine removal efficiencies, as discussed in the Regulatory Guide 1.52 (Ref. 4).

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the PREACS is established by the large break loss of coolant accident (LOCA). The system evaluation assumes a passive failure outside containment, such as a valve packing leakage during a Design Basis Accident (DBA). In such a case, the system restricts the radioactive release to within 10 CFR 100 (Ref. 5) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The analysis of the effects and consequences of a large break LOCA are presented in Reference 3.

There are two types of system failures considered in the accident analysis: a complete loss of function and an excessive LEAKAGE. Either type of failure may result in less efficient removal for any gaseous or particulate material released to the penetration rooms following a LOCA.

The PREACS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant trains of the PREACS are required to be OPERABLE to ensure that at least one train is available, assuming there is a single failure disabling the other train coincident with a loss of offsite power.

The PREACS is considered OPERABLE when the individual components necessary to control radioactive releases are OPERABLE in both trains. A PREACS train is considered OPERABLE when its associated:

a. Fan is OPERABLE,

b. HEPA filter and charcoal absorber are not excessively restricting flow, and are capable of performing the filtration functions, and

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and circulation can be maintained.

The LCO is modified by a Note allowing the penetration room boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for penetration room isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, the PREACS is required to be OPERABLE, consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).

In MODES 5 and 6, the PREACS is not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

BASES

ACTIONS

A.1

With one PREACS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time period, the remaining OPERABLE train is adequate to perform the PREACS function. The 7 day Completion Time is appropriate because the risk contribution of the PREACS is less than that for the ECCS (72 hour Completion Time), and because this system is not a direct support system for the ECCS. The 7 day Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.

B.1

----------REVIEWER'S NOTE----------
Adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of an intentional and unintentional entry into Condition B.

If the penetration room boundary is inoperable, the PREACS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE penetration room boundary within 24 hours. During the period that the penetration room boundary is inoperable, appropriate compensatory measures [consistent with the intent, as applicable, of GDC 19, 60, 64 and 10 CFR Part 100] should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the penetration room boundary.

BASES

ACTIONS (continued)

1

C.1 and C.2

With two PREACS trains inoperable for reasons other than an inoperable boundary, action must be taken to restore at least one PREACS train to OPERABLE status within 24 hours. The Condition is modified by a Note stating it is not applicable if the second PREACS train is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. In addition, at least one train of containment spray must be verified to be OPERABLE within 1 hour. In the event of an accident, containment spray reduces the potential radioactive release from the containment which reduces the consequences of the inoperable PREACS trains. The Completion Time is based on Reference 6 which demonstrated that the 24 hour Completion Time is acceptable based on the infrequent use of the Required Actions and the small incremental effect on plant risk.

D.1 and D.2

----------REVIEWER'S NOTE ----------
Adoption of a MODE 4 end state requires the licensee to make the following commitments:

1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].

2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.

If the inoperable PREACS train or penetration room boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which plant risk is minimized. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

BASES

ACTIONS (continued)

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 7). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Operation [with the heaters on] for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that [heater failure,] blockage, fan or motor failure, or excessive vibration can be detected for corrective action. [The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

1

----------REVIEWER'S NOTE----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

SR 3.7.15.2

This SR verifies the performance of PREACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)]. The [VFTP] includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

[SR 3.7.15.3

This SR verifies that each PREACS train starts and operates on an actual or simulated actuation signal. The SR excludes automatic dampers and valves that are locked, sealed, or otherwise secured in the actuated position. The SR does not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency. [The [18] month Frequency is consistent with that specified in Reference 4.]

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

1

----------REVIEWER'S NOTE----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]]

[SR 3.7.15.4

This SR verifies the integrity of the penetration room enclosure. The ability of the penetration room to maintain negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of the PREACS. During the post accident mode of operation, PREACS is designed to maintain a slightly negative pressure at a flow rate of ≤ [3000] cfm in the penetration room with respect to adjacent areas to prevent unfiltered LEAKAGE.

The minimum system flow rate maintains a slight negative pressure in the penetration room area and provides sufficient air velocity to transport particulate contaminants, assuming only one filter train is operating.

The number of filter elements is selected to limit the flow rate through any individual element to about [1000] cfm. This may vary based on filter housing geometry. The maximum limit ensures that flow through, and pressure drop across, each filter element is not excessive.

The number and depth of the adsorber elements ensures that, at the maximum flow rate, the residence time of the air stream in the charcoal bed achieves the desired adsorption rate. At least a [0.125] second residence time is necessary for an assumed [99]% efficiency.

The filters have a certain pressure drop at the design flow rate when clean. The magnitude of the pressure drop indicates acceptable performance, and is based on manufacturer's recommendations for the filter and adsorber elements at the design flow rate. An increase in pressure drop or decrease in flow indicates that the filter is being loaded or is indicative of other problems with the system.

[The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 8). This test is conducted with the tests for filter penetration; thus, an [18] month Frequency on a STAGGERED TEST BASIS consistent with other filtration SRs.

BASES

SURVEILLANCE REQUIREMENTS (continued)

1

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------REVIEWER'S NOTE----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]]

[SR 3.7.15.5

Operating the PREACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the PREACS filter bypass damper is verified if it can be opened. The SR excludes automatic dampers that are locked, sealed, or otherwise secured in the open position. The SR does not apply to dampers that are locked, sealed, or otherwise secured in the open position since the affected dampers were verified to be in the open position prior to being locked, sealed, or otherwise secured. Placing an automatic damper in a locked, sealed, or otherwise secured position requires an assessment of the OPERABILITY of the system or any supported systems, including whether it is necessary for the damper to be closed to support the accident analysis. Restoration of an automatic damper to the closed position requires verification that the SR has been met within its required Frequency. [An [18] month Frequency is consistent with that specified in Reference 4.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------REVIEWER'S NOTE----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]]

BASES

REFERENCES

1. FSAR, Section [6.5.1].

2. FSAR, Section [9.4.5].

3. FSAR, Section [15.6.5].

4. Regulatory Guide 1.52 Rev. [2].

5. 10 CFR 100.11.

6. WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," Revision 2, August 2010.

7. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

8. NUREG-0800, Section 6.5.1.

1

JUSTIFICATION FOR DEVIATIONS
ISTS 3.7.15, BASES, PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM (PREACS)

1. ISTS 3.7.15, “Penetration Room Exhaust Air Cleanup System (PREACS),” Specification and associated Bases are not included in the PSL ITS because PREACS is not included in the PSL Unit 1 and Unit 2 design.

ENCLOSURE 2

VOLUME 13

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.8
ELECTRICAL SYSTEMS

Revision 1

LIST OF ATTACHMENTS

1. ITS 3.8.1, AC Sources – Operating
2. ITS 3.8.2, AC Sources – Shutdown
3. ITS 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air
4. ITS 3.8.4, DC Sources – Operating
5. ITS 3.8.5, DC Sources – Shutdown
6. ITS 3.8.6, Battery Parameters
7. ITS 3.8.7, Inverters – Operating
8. ITS 3.8.8, Inverters – Shutdown
9. ITS 3.8.9, Distribution Systems – Operating
10. ITS 3.8.10, Distribution Systems – Shutdown

ATTACHMENT 1

ITS 3.8.1, AC Sources - Operating

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.8.1 3.8.1.1 T ~~As a minimum, t~~he following A~~.~~C~~.~~ electrical ~~power~~ sources shall be OPERABLE:

qualified

LCO 3.8.1.a a. Two ~~physically independent~~ circuits between the offsite transmission network and the onsite Class 1E ~~distribution system~~, and

AC Electrical Power Distribution System

LA01

LCO 3.8.1.b b. Two ~~separate and independent~~ diesel generator ~~sets~~ each ~~with~~:

s (DGs)

Day

Insert 1

LA01

SR 3.8.1.4 1. ~~Engine-mounted fuel~~ tanks containing a minimum of 152 gallons of fuel,

2. A separate fuel storage system containing a minimum of 19,000 gallons of fuel, and

~~3. A separate fuel transfer pump.~~

See ITS 3.8.3

A01

Applicability **APPLICABILITY**: MODES 1, 2, 3 and 4.

LA01

ACTION:

ACTION A a. With one offsite circuit ~~of 3.8.1.1.a~~ inoperable~~, except as provided in Action f below:~~

LA02

ACTION A.1 1. P ~~Demonstrate the OPERABILITY of the remaining A.C. sources by p~~erform~~ing~~ S~~urveillance~~ R~~equirement 4.8.1.1.1.a~~ within 1 hour and ~~at least~~ once per 8 hours thereafter.

3.8.1.1

ACTION A.2 2. Within 24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required features(s), declare required features(s) with no offsite power available inoperable when its redundant required features(s) is inoperable.

ACTION A.3 3. Restore the offsite circuit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

ACTION G.1 and G.2

ACTION G.2 NOTE 4. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

A01

INSERT 1

capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.

ELECTRICAL POWER SYSTEMS

ACTION (continued)

~~NOTE~~

~~If the absence of any common-cause failure cannot be confirmed, Surveillance Requirement 4.8.1.1.2.a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.~~

L01

O

b. ~~With o~~ne diesel generator ~~of 3.8.1.1.b~~ inoperable:

P

ACTION B.1

1. ~~Demonstrate the OPERABILITY of the A. C. sources by p~~erform~~ing~~ Surveillance Requirement ~~4.8.1.1.1.~~a within 1 hour and at least once per 8 hours thereafter.

3.8.1.1

ACTION B.2

2. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) ~~(including the steam driven auxiliary feed pump in MODE 1, 2, and 3),~~ declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

LA03

3. ~~If the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing,~~ demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement ~~4.8.1.1.2.a.4~~ within ~~8~~ hours, ~~unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG~~.

LA04

ACTION B.3.2

ACTION B.3.1

3.8.1.2

A01

24

L02

or determine OPERABLE DG is not inoperable due to common cause failure within 24 hours

ACTION B.4

ACTION G

4. Restore the diesel generator to OPERABLE status within 14 days~~*~~ or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

A02

ACTION G.2 NOTE

5. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

~~* A one-time AOT extension for the inoperable 1B EDG allows 30 days to restore the EDG to OPERABLE status. Compensatory Measure within FPL Letter L-2019-153 dated July 25, 2019 will remain in effect during the extended AOT period. This extension expires on August 14, 2019 at 0736 hours EDT.~~

A02

ELECTRICAL POWER SYSTEMS

ACTION (continued)

NOTE

1. Enter applicable ACTIONS of LCO 3.8.2.1, "A.C. Distribution - Operating," when ACTION c is entered with no AC power to any train.
2. ~~If the absence of any common-cause failure cannot be confirmed, Surveillance Requirement 4.8.1.1.2.a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.~~

ACTION D NOTE

L01

ACTION D

c. With one offsite ~~A.C.~~ circuit and one diesel generator inoperable:

ACTION A.1
ACTION B.1

1. Demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement ~~4.8.1.1.1.a~~ 3.8.1.1 within one hour and at least once per 8 hours thereafter.

ACTION B.2

2. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) ~~(including the steam driven auxiliary feed pump in MODE 1, 2, and 3),~~ declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

LA03

3. ~~If the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing,~~ demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 3.8.1.2 ~~4.8.1.1.2.a.4~~ within ~~8~~ 24 hours ~~unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG*~~ or determine OPERABLE DG is not inoperable due to common cause failure within 24 hours.

LA04

L02

A02

ACTION B.3.2

ACTION B.3.1

4. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

L02

ACTIONs D.1 and D.2

ACTION G

ACTION G.2 NOTE

5. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

6. ~~Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 ACTION a or b, as appropriate, with the time requirement of that ACTION based on the time of the initial loss of the remaining inoperable A.C. power source.~~

A03

ACTION C

d. With two ~~of the required~~ offsite A.C. circuits inoperable:

ACTION C.1

1. Within 12 hours from discovery of two offsite circuits inoperable concurrent with inoperability of redundant required feature(s), declare required feature(s) inoperable when its redundant required feature(s) is inoperable.

ACTION C.2

2. Restore one of the inoperable offsite sources to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours.

ATION F.1

3. ~~Following restoration of one offsite source, follow ACTION a with the time requirement of that ACTION based on the time of the initial loss of the remaining inoperable offsite A.C. circuit.~~

A03

ELECTRICAL POWER SYSTEMS

ACTION (continued)

ACTION E

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement ~~4.8.1.1.1.a~~ 3.8.1.1 within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in the at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION B.1

ACTION E

ACTION G

ACTION G NOTE

~~Following restoration of one diesel generator unit, follow ACTION Statement b. with he time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator.~~ A03

With one offsite circuit inoperable

ACTION A

~~f. With one Unit 1 startup transformer (1A or 1B) inoperable and with a Unit 2 startup transformer (2A or 2B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 2 require the use of the startup transformer administratively available to both units,~~ Unit 1 shall demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement ~~4.8.1.1.1.a~~ 3.8.1.1 within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and ~~COLD SHUTDOWN~~ MODE 4 within ~~the following 30~~ 12 hours. LA02 L03 R1

ACTION A.1

ACTION A.3

ACTION G

ACTIONs NOTE

g. LCO 3.0.4.b is not applicable to diesel generators.

Insert proposed ACTION H M01

SURVEILLANCE REQUIREMENTS

SR 3.8.1.1 4.8.1.1.1 Each of the above required ~~independent~~ circuits between the offsite transmission network and the onsite Class 1E distribution system shall be: LA01

SR 3.8.1.1 a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments, indicated power availability; and

SR 3.8.1.8 b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by transferring (manually and automatically) unit power supply from the auxiliary transformer to the startup transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

SR 3.8.1.4 1. Verifying fuel level in the ~~engine-mounted fuel tank~~ day tanks,

2. Verifying the fuel level in the fuel storage tank, See ITS 3.8.3

SR 3.8.1.6 3. Verify~~ing~~ the fuel transfer ~~pump can be started and~~ transfer~~s~~ fuel from the storage ~~system~~ tanks to the ~~engine-mounted tank~~ day tanks, oil system operates to automatically M02

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.2
SR 3.8.1.7

4. Verifying ~~the~~ each diesel starts from ~~ambient~~ standby condition ~~and accelerates to approximately 900 rpm~~ in less than or equal to 10 seconds**. The generator voltage and frequency shall be ≥ 3950V and ≥ 59.4 Hz within 10 seconds and achieve a steady state voltage and frequency of 4160 ± ~~420~~ 210 volts and 60 ± ~~1.2~~ 0.6 Hz ~~within 10 seconds after the start signal~~**. ~~The diesel generator shall be started for this test by using one of the following signals:~~ L04 L04

~~a) Manual/Local~~

~~b) Simulated loss-of-offsite power by itself.~~ L05

~~c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.~~

~~d) An ESF actuation test signal by itself.~~ A04 M03 M04

Add proposed SR 3.8.1.3 Note 1
Add proposed SR 3.8.1.3 Note 3
Add proposed SR 3.8.1.3 Note 4
and

SR 3.8.1.3
SR 3.8.1.3 NOTE 2

5. Verifying the generator is synchronized~~,~~ loaded ~~to greater than or equal to 3500 kW~~ ≥ in accordance with the manufacturer's recommendations and operates within a load band of 3300 to 3500 kW*** for ~~at least an additional~~ 60 minutes~~, and~~ L06

~~6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.~~ LA05

b. ~~By removing~~ Check for and remove accumulated water

SR 3.8.1.5

1. From ~~the engine-mounted fuel~~ each day tank in accordance with the Surveillance Frequency Control Program ~~and after each occasion when the diesel is operated for greater than 1 hour, and~~ L07

2. From the storage tank in accordance with the Surveillance Frequency Control Program.

See ITS 3.8.3

SR 3.8.1.2
SR 3.8.1.7

** The diesel generator start (10 sec.) from ambient conditions shall be performed in accordance with the Surveillance Frequency Control Program in these surveillance tests. All other diesel generator starts for the purposes of this surveillance testing may be preceded by an engine prelube period and may also include warmup procedures (e.g., gradual acceleration) as recommended by the manufacturer ~~so that mechanical stress and wear on the diesel generator is minimized.~~ A05

SR 3.8.1.2
NOTEs 1 and 2
SR 3.8.1.7
NOTE

SR 3.8.1.3
NOTE 2

*** ~~The indicated load band is meant as guidance to avoid routine overloading.~~ Variations in loads in excess of the band due to changing bus loads shall not invalidate this test. A06

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

c. Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of the Diesel Fuel Oil Testing Program.

See ITS 3.8.3

SR 3.8.1.9
SR 3.8.1.10
SR 3.8.1.11
SR 3.8.1.12
SR 3.8.1.13
SR 3.8.1.14
SR 3.8.1.15
SR 3.8.1.16
SR 3.8.1.17

~~d. DELETED~~

e. In accordance with the Surveillance Frequency Control Program by:

~~1. DELETED~~

2.

Insert proposed SR 3.8.1.9 NOTE 2

M05

SR 3.8.1.9
NOTE 1, Part 2

Insert 2

NOTE

Credit may be taken for unplanned events that satisfy this SR.

Insert proposed SR 3.8.1.9.a

each

SR 3.8.1.9.b and c

Verify~~ing~~ generator ~~capability to~~ reject ~~the~~ single largest post-accident load ~~while maintaining~~ voltage at 4160 ± ~~420~~ volts and frequency at 60 ± ~~1.2~~ Hz.

0.6

210

s a load greater than or equal to its associated

L08

3.

and within 1.8 seconds following load rejection

SR 3.8.1.10
NOTE 2

NOTE

This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.10

Verifying that upon an actual or simulated loss of offsite power signal ~~by itself~~.

SR 3.8.1.10.a

a) Deenergization of the emergency busses and load shedding from the emergency busses.

SR 3.8.1.10.b

M05

INSERT 2

1. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.1.10.c

b) The diesel starts on the auto-start signal****, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer timers and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 210 volts and 60 ± 0.6 Hz during this test.

4.

SR 3.8.1.11 NOTE 2

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.11

Verifying that upon an actual or simulated ESF actuation signal ~~(without loss-of-offsite power)~~ the diesel generator starts**** on the auto-start signal, and: A07

SR 3.8.1.11.a

a) Within 10 seconds, generator voltage and frequency shall be ~~4160 ± 420 volts and 60 ± 1.2 Hz~~ ≥ 3950 V and ≥ 59.4 Hz L04

SR 3.8.1.11.c

b) Operates on standby for greater than or equal to 5 minutes.

SR 3.8.1.11.b

c) Steady-state generator voltage and frequency shall be 4160 ± 210 volts and 60 ± 0.6 Hz ~~and shall be maintained throughout this test~~. M06

Proposed SR 3.8.1.11 Requirements d and e.

5.

SR 3.8.1.17 NOTE 2
SR 3.8.1.12 NOTE

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.17

Verifying that upon an actual or simulated loss-of-offsite power signal in conjunction with an ESF actuation signal: actual or simulated L09

SR 3.8.1.17.a
SR 3.8.1.17.b

a) Deenergization of the emergency busses and load shedding from the emergency busses.

SR 3.8.1.10 NOTE 1
SR 3.8.1.11 NOTE 1

preceded by an

**** This test may be ~~conducted in accordance with the manufacturer's recommendations concerning~~ engine prelube period. LA01

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.1.17

b) The diesel starts on the auto-start signal****, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency ~~(accident)~~ loads through the ~~auto-sequencer~~ and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 210 volts and 60 ± 0.6 Hz during this test.

load sequence timers

SR 3.8.1.12

c) ~~All automatic~~ diesel generator trips~~, except engine overspeed and generator differential,~~ are ~~automatically~~ bypassed ~~upon~~ loss of voltage on the emergency bus concurrent with a ~~safety injection~~ signal.

Verify each

's noncritical automatic

signal

an actual or simulated ESF actuation

actual or simulated

LA06

L09

M05

6. Add proposed SR 3.8.1.13 Note 3

SR 3.8.1.13 Note 2

NOTE
Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.13

each

≥

Verify~~ing the~~ diesel generator operates for ~~at least~~ 24 hours****. During the first 2 hours ~~of this test~~, the diesel generator shall be loaded within a load band of 3800 to ~~3960~~ kW# and during the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3300 to 3500 kW#. ~~The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.~~

3860

A08

M09

L10

~~7. DELETED~~

8.

SR 3.8.1.14 Note

NOTE
This Surveillance shall not normally be performed in MODE 1, 2, 3 or 4. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

Verifying the diesel generator's capability to:

SR 3.8.1.14

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon actual or simulated restoration of offsite power.

b) Transfer its load to the offsite power source, and

c) Be restored to its standby status.

SR 3.8.1.13 Note 1

\# ~~This band is meant as guidance to avoid routine overloading of the engine.~~ Variations in load in excess of this band due to changing bus loads shall not invalidate this test.

A06

SR 3.8.1.17 NOTE 1

**** This test may be ~~conducted in accordance with the manufacturer's recommendations concerning~~ engine prelube period.

preceded by an

LA01

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

9.

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.15 NOTE

Verify~~ing that~~ with the diesel generator operating in a test mode ~~(~~connected to its bus~~)~~, an actual or simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

SR 3.8.1.15

~~10. DELETED~~

11.

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.16 NOTE

SR 3.8.1.16

sequenced

Verify~~ing that the~~ automatic load sequence timers are operable with the interval between each load block within ± 1 second of its design interval.

SR 3.8.1.18

f. In accordance with the Surveillance Frequency Control Program ~~or after any modification which could affect diesel generator independence~~ by starting**** the diesel generators simultaneously~~, during shutdown,~~ and verifying that the diesel generators ~~accelerate to approximately 900 rpm~~ in less than or equal to 10 seconds.

L11 L12 L04 M07

achieves

Insert proposed SR 3.8.1.18.a and b

SR 3.8.1.18 NOTE

~~g. In accordance with the Surveillance Frequency Control Program by performing a pressure test of those portions of the diesel fuel oil system designed to USAS B31.7 Class 3 requirements in accordance with the Inservice Inspection Program.~~

LA07

~~4.8.1.1.3 Reports – (Not Used)~~

~~4.8.1.1.4 The Class 1E underground cable system shall be demonstrated OPERABLE within 30 days after the movement of any loads in excess of 80% of the ground surface design basis load over the cable ducts by pulling a mandrel with a diameter of at least 80% of the duct's inside diameter through a duct exposed to the maximum loading (duct nearest the ground's surface) and verifying that the duct has not been damaged.~~

LA08

~~# This band is meant as guidance to avoid routine overloading of the engine. Variations in load in excess of this band due to changing bus loads shall not invalidate this test.~~

A09

SR 3.8.1.18 NOTE

**** This test may be ~~conducted in accordance with the manufacturer's recommendations concerning~~ engine prelube period.

LA01

preceded by an

~~TABLE 3.4 8-1~~

~~DIESEL GENERATOR TEST SCHEDULE~~

~~(NOT USED)~~

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.8.1 3.8.1.1 ~~As a minimum,~~ the following A~~.~~C~~.~~ electrical ~~power~~ sources shall be OPERABLE:

a. Two ~~physically independent~~ circuits between the offsite transmission network and the onsite Class 1E ~~distribution system~~, and

b. Two ~~separate and independent~~ diesel generators, each ~~with~~:

1. Two separate engine-mounted fuel tanks containing a minimum volume of 238 gallons of fuel each,

2. A separate fuel storage system containing a minimum volume of 42,500 gallons of fuel, and

3. A separate fuel transfer pump.

Applicability **APPLICABILITY:** MODES 1, 2, 3, and 4.

ACTION:

ACTION A a. With one offsite circuit ~~of 3.8.1.1.a~~ inoperable~~, except as provided in ACTION f below:~~

ACTION A.1 1. ~~Demonstrate the OPERABILITY of the remaining A. C. sources by p~~erform~~ing~~ S~~urveillance~~ R~~equirement~~ 4.8.1.1.1.a within 1 hour and ~~at least~~ once per 8 hours thereafter.

ACTION A.2 2. Within 24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s), declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.

ACTION A.3 3. Restore the offsite circuit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

ACTION F.1 and F.2

ACTION F.2 NOTE 4. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

A01

INSERT 1

capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System.

ELECTRICAL POWER SYSTEMS

ACTION (continued)

~~**NOTE**~~

~~If the absence of any common-cause failure cannot be confirmed, Surveillance requirement 4.8.1.1.2.a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.~~

L01

O

b. ~~With o~~ne diesel generator ~~of 3.8.1.1.b~~ inoperable:

P

ACTION B.1

1. ~~Demonstrate the OPERABILITY of the A.C. sources by p~~erform~~ing~~ Surveillance Requirement ~~4.8.1.1.1.a~~ within 1 hour and at least once per 8 hours thereafter.

3.8.1.1

ACTION B.2

2. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) ~~(including the steam driven auxiliary feed pump in MODE 1, 2, and 3),~~ declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

LA03

LA04

3. ~~If the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing,~~ demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 24

L02

ACTION B.3.2

3.8.1.2

~~4.8.1.1.2.a.4~~ within ~~8~~ hours, ~~unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG~~.

A01

ACTION B.3.1

or determine OPERABLE DG is not inoperable due to common cause failure within 24 hours

ACTION B.4

4. Restore the diesel generator to OPERABLE status within 14 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION G

ACTION G.2 NOTE

5. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ELECTRICAL POWER SYSTEMS

ACTION: (Continued)

NOTE

- ~~If the absence of any common cause failure cannot be confirmed, Surveillance Requirement 4.8.1.1.2.a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.~~ L01
- Enter applicable ACTIONS of LCO 3.8.3.1, "Onsite Power Distribution – Operating," when ACTION c is entered with no AC power to any train. (ACTION D NOTE)

ACTION D

c. With one offsite A.C. circuit and one diesel generator inoperable:

1. Demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement ~~4.8.1.1.1.a~~ 3.8.1.1 within one hour and at least once per 8 hours thereafter. (ACTION A.1, ACTION B.1)

2. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) ~~(including the steam driven auxiliary feed pump in MODE 1, 2, and 3),~~ declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable. (ACTION B.2) LA03

3. ~~If the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing,~~ demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement ~~4.8.1.1.2.a.4~~ 3.8.1.2 within ~~8~~ 24 hours, ~~unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG.~~ or determine OPERABLE DG is not inoperable due to common cause failure within 24 hours (ACTION B.3.2, ACTION B.3.1) LA04, L02

4. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. (ACTIONs D.1 and D.2, ACTION G)

5. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN. (ACTION G.2 NOTE)

6. ~~Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 ACTION Statement a or b, as appropriate, with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable A.C. power source.~~ A03

ELECTRICAL POWER SYSTEMS

ACTION: (Continued)

ACTION C

d. With two ~~of the required~~ offsite A.C. circuits inoperable:

ACTION C.1

1. Within 12 hours from discovery of two offsite circuits inoperable concurrent with inoperability of redundant required feature(s), declare required feature(s) inoperable when its redundant required feature(s) is inoperable.

ACTION C.2

ACTION F.1

2. Restore one of the inoperable offsite sources to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours.

3. ~~Following restoration of one offsite source, follow ACTION a with the time requirement of that ACTION based on the time of the initial loss of the remaining inoperable offsite A.C. circuit.~~ A03

ACTION E

e. With two of the above required diesel generators inoperable, demonstrate the

ACTION B.1

OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement

3.8.1.1

~~4.8.1.1.1.a within~~ 1 hour and at least once per 8 hours thereafter; restore one of

ACTION E

the inoperable diesel generators to OPERABLE status within 2 hours or be in the at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within

ACTION G

the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT

ACTION G NOTE

SHUTDOWN. ~~Following restoration of one diesel generator unit, follow ACTION Statement b. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator.~~ A03

With one offsite circuit inoperable

ACTION A

~~f.~~ ~~With one Unit 2 startup transformer (2A or 2B) inoperable and with a Unit 1 startup transformer (1A or 1B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 1 require the use of the startup transformer administratively available to both units,~~ Unit 2 shall LA02

3.8.1.1

ACTION A.1

demonstrate the operability of the remaining A.C. sources by performing Surveillance Requirement ~~4.8.1.1.1a~~. within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within

ACTION A.3

72 hours or in accordance with the Risk Informed Completion Time Program, or be

ACTION G

in at least HOT STANDBY within the next 6 hours and ~~COLD SHUTDOWN~~ within ~~the following 30~~ 12 hours. R1

MODE 4

L03

ACTIONs NOTE

g. LCO 3.0.4.b is not applicable to diesel generators.

Insert proposed ACTION H M01

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required ~~independent~~ circuits between the offsite transmission network and the onsite Class 1E distribution system shall be: LA01

SR 3.8.1.1

a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments, indicated power availability; and

SR 3.8.1.8

b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.1.4

1. Verifying fuel level in the ~~engine-mounted fuel tank~~, day tanks

2. Verifying the fuel level in the fuel storage tank, See ITS 3.8.3

SR 3.8.1.6

3. Verify~~ing~~ the fuel oil transfer ~~pump can be started and~~ transfer~~s~~ fuel from the storage ~~system~~ to the ~~engine-mounted tank~~, system operates to automatically M02

SR 3.8.1.2
SR 3.8.1.7

4. Verifying ~~the~~ diesel starts from ~~ambient~~ condition ~~and accelerates to approximately 900 rpm~~ in less than or equal to 10 seconds**. The generator voltage and frequency shall be 4160 ± ~~420~~ volts and 60 ± ~~1.2~~ Hz ~~within 10 seconds after the start signal~~**. ~~The diesel generator shall be started for this test by using one of the following signals:~~ L04, 0.6, 210, ≥ 3950V and ≥ 59.4 Hz within 10 seconds and achieve a steady state voltage and frequency of, L04

~~a) Manual/Local.~~

~~b) Simulated loss-of-offsite power by itself.~~

~~c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.~~ L05

~~d) An ESF actuation test signal by itself.~~ A04, Add proposed SR 3.8.1.3 Note 1, M03, Add proposed SR 3.8.1.3 Note 3, M04, Add proposed SR 3.8.1.3 Note 4

SR 3.8.1.3
SR 3.8.1.3
NOTE 2

5. Verifying the generator is synchronized~~,~~ and loaded ~~to greater than or equal to 3685 kW~~ in accordance with the manufacturer's recommendations, and operates within a load band of 3450 to ~~3685~~ 3669 kW*** for ~~at least an additional~~ ≥ 60 minutes~~, and~~ L06, M10

~~6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.~~ LA05

b. ~~By removing~~ Check for and remove accumulated water:

SR 3.8.1.5

1. From ~~the engine-mounted fuel~~ each day tank in accordance with the Surveillance Frequency Control Program ~~and after each occasion when the diesel is operated for greater than 1 hour, and~~ L07

2. From the storage tank in accordance with the Surveillance Frequency Control Program. See ITS 3.8.3

SR 3.8.1.2 **
SR 3.8.1.7

The diesel generator start (10 sec.) from ambient conditions shall be performed in accordance with the Surveillance Frequency Control Program in these surveillance tests. All other diesel generator starts for purposes of this surveillance testing may be preceded by an engine prelube period and may also include warmup procedures (e.g., gradual acceleration) as recommended by the manufacturer ~~so that mechanical stress and wear on the diesel generator is minimized~~.

SR 3.8.1.2
NOTEs 1 and 2
SR 3.8.1.7
NOTE

A05

SR 3.8.1.3
NOTE 2

*** ~~The indicated load band is meant as guidance to avoid routine overloading.~~ Variations in loads in excess of the band due to changing bus loads shall not invalidate this test. A06

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9
SR 3.8.1.10
SR 3.8.1.11
SR 3.8.1.12
SR 3.8.1.13
SR 3.8.1.14
SR 3.8.1.15
SR 3.8.1.16
SR 3.8.1.17
SR 3.8.1.19
SR 3.8.1.20

c. Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of the Diesel Fuel Oil Testing Program.

See ITS 3.8.3

~~d. DELETED~~

e. In accordance with the Surveillance Frequency Control Program by:

~~1. DELETED~~

2.

SR 3.8.1.9 NOTE 1, Part 2

Insert proposed SR 3.8.1.9.a

Insert proposed SR 3.8.1.9 NOTE 2

M05

NOTE

Insert 2

Credit may be taken for unplanned events that satisfy this SR.

each

SR 3.8.1.9.b and c

Verify~~ing~~ generator ~~capability to~~ reject ~~the~~ single largest post-accident load ~~while maintaining~~ voltage at 4160 ± ~~420~~ volts and frequency at 60 ± ~~1.2~~ Hz.

210

0.6

s a load greater than or equal to its associated

L08

3.

SR 3.8.1.19 NOTE 1. Part 2

and within 1.8 seconds following load rejection

Insert proposed SR 3.8.1.19 NOTE 2

M08

NOTE

Insert 2

Credit may be taken for unplanned events that satisfy this SR.

each DG does not trip, and voltage is maintained ≤

SR 3.8.1.19

Verify~~ing the generator capability to reject a load of 3685 kW without tripping. The generator voltage shall not exceed~~ 4784 volts during and following ~~the~~ load rejection.

a

of ≥ 3669 kW

M10

M05 M08

INSERT 2

1. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.10 NOTE 2

SR 3.8.1.10 Verifying upon an actual or simulated loss-of-offsite power signal ~~by itself~~:

SR 3.8.1.10.a SR 3.8.1.10.b a. Deenergization of the emergency busses and load shedding from the emergency busses.

b. The diesel starts on the auto-start signal,**** energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer~~r~~ timers and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 210 volts and 60 ± 0.6 Hz during this test.

SR 3.8.1.10.c

5.

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.11 NOTE 2

SR 3.8.1.11 Verifying that upon an actual or simulated ESF actuation signal ~~(without loss-of-offsite power)~~ the diesel generator starts**** on the auto-start signal, and: A07

SR 3.8.1.11.a a) Within 10 seconds, generator voltage and frequency shall be ~~4160 ± 420 volts and 60 ± 1.2 Hz~~ ≥ 3950 V and ≥ 59.4 L04

SR 3.8.1.11.c b) Operates on standby for greater than or equal to 5 minutes.

SR 3.8.1.11.b c) Steady-state generator voltage and frequency shall be 4160 ± 210 volts and 60 ± 0.6 Hz and shall be maintained throughout this test.

Insert proposed SR 3.8.1.11 Requirements d and e. M06

SR 3.8.1.10 NOTE 1 SR 3.8.1.11 NOTE 1

**** This test may be ~~conducted in accordance with the manufacturer's recommendations concerning~~ preceded by an engine prelube period. LA01

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

6.

SR 3.8.1.12 NOTE
SR 3.8.1.17 NOTE 2

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.17

Verifying that upon an actual or simulated loss-of-offsite power in conjunction with an ESF actuation signal: (actual or simulated) L09

SR 3.8.1.17.a
SR 3.8.1.17.b

a) Deenergization of the emergency busses and load shedding from the emergency busses.

b) The diesel starts on the auto-start signal,**** energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer (timers) and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 210 volts and 60 ± 0.6 Hz during this test.

SR 3.8.1.17

SR 3.8.1.12

c) Verify~~ing that all automatic~~ (each) diesel generator ('s noncritical automatic) trips~~, except engine overspeed and generator differential,~~ are ~~automatically~~ bypassed ~~upon~~ loss of voltage on the emergency bus concurrent with ~~a safety injection~~ (an actual or simulated ESF) actuation signal. (signal) (actual or simulated) LA06 L09

7.

SR 3.8.1.13 Note 2

Add proposed SR 3.8.1.13 Note 3

NOTE
Credit may be taken for unplanned events that satisfy this SR. M05

SR 3.8.1.13

Verify~~ing the~~ (each) diesel generator operates for ~~at least~~ (≥) 24 hours.**** During the first 2 hours of this test, the diesel generator shall be loaded within a load band of 3800 to ~~3985~~ (3934) kW# and during the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3450 to ~~3685~~ (3669) kW#. ~~The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test.~~ A08 M09 M10 L10

~~8. DELETED~~

SR 3.8.1.13 NOTE 1 # ~~This band is meant as guidance to avoid routine overloading of the engine.~~ Variations in load in excess of this band due to changing bus loads shall not invalidate this test. A06

SR 3.8.1.17 NOTE 1 **** This test may be ~~conducted in accordance with the manufacturer's recommendations concerning~~ (preceded by an) engine prelube period. LA01

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

9.

SR 3.8.1.14 Note

NOTE
This Surveillance shall not normally be performed in MODE 1, 2, 3 or 4. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

Verifying the diesel generator's capability to:

SR 3.8.1.14

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon actual or simulated restoration of offsite power signal.

b) Transfer its load to the offsite power source, and

c) Be restored to its standby status.

10.

SR 3.8.1.15 NOTE

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.15

Verify~~ing that~~ with the diesel generator operating in a test mode ~~(~~connected to its bus~~)~~, an actual or simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

~~11. DELETED~~

12.

SR 3.8.1.16 NOTE

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.16

Verify~~ing that the~~ automatic load sequence timers are operable with the interval between each sequenced load block within ±1 second of its design interval.

SR 3.8.1.20

13. Performing ~~Surveillance Requirement 4.8.1.1.2a.4~~ within 5 minutes of a DG start shutting down the diesel generator after it has operated within a load band of 3450 kW to ~~3685~~ 3669 kW# for at least 2 hours or until operating temperatures have stabilized. A10 M10

SR 3.8.1.20 NOTE 1, Part 1

SR 3.8.1.20 NOTE 1, Part 2

~~This band is meant as guidance to avoid routine overloading of the engine.~~ Variations in load in excess of this band due to changing bus loads shall not invalidate this test. A06

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.1.18

f. In accordance with the Surveillance Frequency Control Program ~~or after any modifications which could affect diesel generator interdependence~~ by starting**** the diesel generators simultaneously~~, during shutdown,~~ and verifying that the diesel generators ~~accelerate to approximately 900 rpm~~ in less than or equal to 10 seconds.

achieves

Insert proposed SR 3.8.1.18.a and b

L11

L12

L04

M07

g. ~~In accordance with the Surveillance Frequency Control Program by performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with the Inservice Inspection Program.~~

LA07

~~4.8.1.1.3 Reports – (Not Used).~~

~~4.8.1.1.4 The Class 1E underground cable system shall be demonstrated OPERABLE within 30 days after the movement of any loads in excess of 80% of the ground surface design basis load over the cable ducts by pulling a mandrel with a diameter of at least 80% of the duct's inside diameter through a duct exposed to the maximum loading (duct nearest the ground's surface) and verifying that the duct has not been damaged.~~

LA08

SR 3.8.1.18 NOTE

**** This test may be ~~conducted in accordance with the manufacturer's recommendations concerning~~ engine prelube period.

preceded by an

LA01

~~TABLE 4.8-1~~

~~DIESEL GENERATOR TEST SCHEDULE~~

~~(NOT USED)~~

DISCUSSION OF CHANGES
ITS 3.8.1, AC SOURCES - OPERATING

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 1 only:** CTS 3.8.1.1 Actions b.4 and c.3 contain footnote * that applies a "one time extension for the inoperable 1B EDG" to allow 30 days to restore the diesel generator to OPERABLE status. The extension expired on August 14, 2019. Footnote * is deleted and not included in ITS. The change is designated as administrative and does not result in technical changes to the CTS.

A03 CTS 3.8.1.1 Actions c.6, d.3, and last sentence of Action e, each provide requirements to follow other CTS 3.8.1.1 Actions upon restoration of the inoperable equipment. The ITS provide these requirements in Section 1.3, Completion Times. This changes the CTS by removing the requirements on application of Completion Times from individual CTS Actions to the use and application chapter in ITS.

The purpose of the CTS requirement to follow other actions upon restoration of the inoperable equipment is to provide direction, when multiple components are inoperable, on the appropriate application of action completion times when one component remains inoperable following the restoration of other required equipment. These CTS requirements were added as a result of adopting portions of the ISTS without fully converting to the ISTS.

ITS includes Section 1.3, "Completion Times," which provides requirements on the proper use and application of Completion Times when components remain inoperable following restoration of other equipment. Therefore, it is unnecessary to duplicate the requirements in individual Specifications. This change is designated as administrative because the CTS requirements are consistent with the Completion Time requirements in ITS Section 1.3 and does not result in technical changes to the CTS.

A04 Unit 1 CTS 4.8.1.1.2.a.5 requires that the DG is synchronized, loaded to greater than or equal to 3500 kW in accordance with the manufacturer's recommendations and operates within a load band of 3300 to 3500 kW for at least an additional 60 minutes. Unit 2 CTS 4.8.1.1.2.a.5 requires that the DG is synchronized, loaded to greater than or equal to 3685 kW in accordance with the manufacturer's recommendations, and operates within a load band of 3450 to 3685 kW for at least an additional 60 minutes. ITS SR 3.8.1.3 is modified by Note 1 that states DG loadings may include gradual loadings as recommended by the manufacturer. This changes the CTS by allowing the DG loading to be gradual "as recommended by the manufacturer."

The purpose of CTS 4.8.1.1.2.a.5 is to ensure each DG can be loaded from standby conditions and operated at load for at least one hour. The change to CTS 4.8.1.1.2.a.5 includes the intent provided by the manufacturer to continue the DG warmup by gradually loading the DG. This change is consistent with current practice and is not precluded by CTS. This change is consistent with the recommendations of Generic Letter (GL) 93-05, "Line- Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation." GL 93-05, Section 10.1, states that DGs "should be loaded in accordance with vendor recommendations for all test purposes other than the refueling outage loss of offsite power tests." The change is acceptable, because it will ensure the DGs will continue to be operated consistent with manufacturer recommendations. This change is designated as administrative, because it does not result in a technical change to the CTS.

A05 CTS 4.8.1.1.2.a.4 footnote ** includes the phrase, “so that mechanical stress and wear on the diesel generator is minimized” in addressing the allowed use of manufacturer recommended procedures. ITS does not include this additional clarifying information. Deleting this text does not change the allowance of engine prelube periods and warmup procedures as recommended by the manufacturer for performance of the applicable surveillance requirements. | R1

The purpose of the phrase “minimizing mechanical stress and wear on the diesel generator” is to provide additional clarification for the allowance. This change is designated as administrative because it does not result in technical changes to the CTS.

A06 CTS 4.8.1.1.2.a.5 footnote *** includes the statement, “The indicated load band is meant as guidance to avoid routine overloading.” Unit 1 CTS 4.8.1.1.2.e.6, Unit 2 CTS 4.8.1.1.2.e.7 and Unit 2 CTS 4.8.1.1.2.e.13 footnote # includes the statement, “This band is meant as guidance to avoid routine overloading of the engine.” ITS does not include this additional clarifying information. Deleting this text does not change the allowance for variations in loads outside of the stated band due to changing bus loads when meeting the surveillance criteria. | R1

The purpose of the statements is clarification of the load band in justifying the allowance of variations in loads due to bus load changes during performance of the surveillance. This change is designated as administrative because it does not result in technical changes to the CTS.

A07 Unit 1 CTS 4.8.1.1.2.e.4 and Unit 2 CTS 4.8.1.1.2.e.5 require verification that on an ESF actuation signal (without loss of offsite power), each DG starts and operates for greater than or equal to 5 minutes. ITS SR 3.8.1.11 require a similar test but do not specify that the DG auto-start on an ESF actuation signal is "without loss of offsite power." This changes the CTS by not specifying the DG auto-start on an ESF actuation signal is without a loss of offsite power signal.

The purpose of Unit 1 CTS 4.8.1.1.2.e.4 and Unit 2 CTS 4.8.1.1.2.e.5 is to demonstrate that each DG automatically starts on an actual or simulated ESF actuation signal. The requirements of this Surveillance are retained in the ITS as SR 3.8.1.11. This change is acceptable, because it is understood that the CTS Unit 1 4.8.1.1.2.e.4 and Unit 2 CTS 4.8.1.1.2.e.5 required DG start on an ESF

actuation signal is without a loss of offsite power signal. Therefore, it is unnecessary to provide this information in ITS SR 3.8.1.11. This change is designated as administrative because it does not result in a technical change to the CTS.

A08 Unit 1 CTS 4.8.1.1.2.e.6 and Unit 2 CTS 4.8.1.1.2.e.7 require the DG to operate for at least 24 hours and contains a **** footnote indicating the test may be conducted in accordance with manufacturer's recommendation concerning engine prelube period. ITS SR 3.8.1.13 does not retain this note because the note is related to DG start tests and not associated with the DG 24 hour load run test. ITS Surveillances related to DG starting contain a similar note allowing engine prelube prior to the DG start. This change is designated as administrative because it does not result in a technical change to the CTS.

A09 **Unit 1 only**: CTS page 3/4.8-6b contains a # footnote not associated with any requirement on the page. ITS does not retain this footnote. The change is designated as administrative and does not result in technical changes to the CTS.

A10 **Unit 2 only**: CTS 4.8.1.1.2.e.13 requires performing Surveillance Requirement 4.8.1.1.2a.4 within 5 minutes of shutting down the diesel generator following a load run (i.e., DG hot restart test). ITS SR 3.8.1.20 retains this requirement as Note 2 to the Surveillance and instead of referencing another Surveillance to perform, ITS SR 3.8.1.20 explicitly provides the DG start requirements. Refer to DOC L04 for changes to CTS 4.8.1.1.2.a.4. The change is designated as administrative because explicitly stating the requirements of a Surveillance instead of referencing the Surveillance does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 The St. Lucie Plant (PSL) CTS do not contain a Condition for three or more AC sources inoperable. ITS ACTION H requires immediate entry into LCO 3.0.3 when three or more AC sources are inoperable. This changes the CTS by adding a specific Action requiring entry into LCO 3.0.3 when three or more AC sources are inoperable.

The addition of ITS ACTION is acceptable because it addresses a level of degradation in which all redundancy in AC electrical power supplies has been lost and any further losses in the AC electrical power system will result in loss of safety function. This change is designated as more restrictive because additional requirements are being imposed that limit unit operation in the event of 3 or more AC sources inoperable.

M02 CTS 4.8.1.1.2.a.3 requires verification that the fuel transfer pump can be started and transfer fuel from the storage system to the day tank. ITS SR 3.8.1.6 requires verification the fuel oil transfer system operates to automatically transfer fuel oil from the storage tanks to the day tanks. This changes the CTS requirement from verifying that the fuel oil transfer pump can be started and transfer fuel from the storage system to requiring verification that the fuel oil

transfer system operates to automatically transfer fuel oil from the storage tanks to the day tanks.

The purpose of CTS 4.8.1.1.2.a.3 is to demonstrate that each fuel oil transfer pump operates and transfers fuel oil from the storage tanks to the day tanks. The proposed change demonstrates that the fuel oil transfer system operates to automatically transfer fuel to the day tank of each DG engine to support continuous operation of the standby power source. The change is acceptable because the fuel transfer system consists of a transfer pump, day tank isolation valves and day tank level switches that operate in conjunction to automatically maintain DG engine day tank level in standby conditions and also ensure adequate fuel supply to an operating diesel generator for continuous operation. In addition, this change demonstrates that the controls and control system for automatic fuel transfer is OPERABLE. This change is designated as more restrictive because it requires demonstration of automatic operation of the fuel oil transfer system.

M03 CTS 4.8.1.1.2.a.5 requires each DG to be synchronized, loaded, and operated for at least 60 minutes. ITS SR 3.8.1.3 requires the same test however an additional Note has been added that places a restriction on the test. ITS SR 3.8.1.3 Note 3 modifies the CTS requirements by stating that the SR shall be on only one DG at a time. This changes the CTS by adding a restriction when performing this test.

The purpose of CTS 4.8.1.1.2.a.5 is to ensure the DG can supply the emergency loads. This change is acceptable because DG loading during performance of CTS 4.8.1.1.2.a.5 is usually conducted one DG at a time. This change is designated as more restrictive because an explicit restriction is added to the DG load test.

M04 CTS 4.8.1.1.2.a.5 requires each DG to be synchronized, loaded, and operated for at least 60 minutes. ITS SR 3.8.1.3 requires the same test however an additional Note has been added that places a restriction on the test. ITS SR 3.8.1.3 Note 4 modifies the CTS requirements by stating that the SR shall be preceded by and immediately follow, without a shutdown of the DG, a successful performance of ITS SR 3.8.1.2 or ITS SR 3.8.1.7. This changes the CTS by adding a restriction when performing this test.

The purpose of CTS 4.8.1.1.2.a.5 is to ensure the DG can supply the emergency loads. This change is acceptable because DG loading during performance of CTS 4.8.1.1.2.a.5 is usually conducted without shut down after a successful start during performance of CTS 4.8.1.1.2.a.4. This change is designated as more restrictive because an explicit restriction is added to the DG load test.

M05 Unit 1 CTS 4.8.1.1.2.e.2, Unit 1 CTS 4.8.1.1.2.e.6, Unit 2 CTS 4.8.1.1.2.e.2 and Unit 2 CTS 4.8.1.1.2.e.7 require the testing of each DG with a load rejection greater than or equal to the single largest post-accident load. The respective CTS requires verifying the diesel generator operates for at least 24 hours under specific loaded conditions. The Surveillances do not specify that the testing be performed under certain unit conditions, nor that a DG shall be tested at a specific power factor. ITS SR 3.8.1.9 requires the verification that each DG can

reject a load equal to or greater than its associated single largest post-accident load. Unit 1 ITS SR 3.8.1.13 requires verifying each DG operates for ≥ 24 hours; for ≥ 2 hours loaded ≥ 3800 kW and ≤ 3860 kW and for the remaining hours of the test loaded ≥ 3300 kW and ≤ 3500 kW. Unit 2 ITS SR 3.8.1.13 requires verifying each DG operated for ≥ 24 hours; for ≥ 2 hours loaded ≥ 3800 kW and ≤ 3934 kW and for the remaining hours of the test loaded ≥ 3450 kW and ≤ 3669 kW. Two Notes modify these SRs. ITS SR 3.8.1.9 Note 1 states that for the associated unit's DGs, the Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. ITS SR 3.8.1.9 Note 2 and Unit 1 ITS SR 3.8.1.13 and Unit 2 ITS 3.8.1.13 Note 3 state that if the Surveillance is performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. This changes the CTS requirement by specifying conditions for when the associated unit DGs are normally tested and by specifying a power factor of ≤ 0.9 if the testing is conducted by synchronizing with the offsite sources. Other changes to Unit 1 and Unit 2 CTS 4.8.1.1.2.e.2 (ITS SR 3.8.1.9), Unit 1 CTS 4.8.1.1.2.e.6 (ITS 3.8.1.13) and Unit 2 CTS 4.8.1.1.2.e.7 (ITS 3.8.1.13) are discussed in DOC L08 and DOC L10.

The addition of ITS SR 3.8.1.9 Notes 1 and 2, ITS SR 3.8.1.13 Note 3 is acceptable, because performance of the Surveillance for the associated unit's DGs could cause perturbations to the electrical distribution systems. Restricting normal performance of the Surveillance on the associated unit's DGs during MODES 1 and 2 minimizes challenges to continued steady state operation and, as a result, the unit's safety systems. However, if an assessment of the risks indicates plant safety is maintained or enhanced, the Surveillance can be performed to reestablish OPERABILITY. The addition of ITS SR 3.8.1.9 Note 2, Unit 1 ITS SR 3.8.1.13 Notes 2 Note 3, and Unit 2 ITS SR 3.8.1.14 Notes 2 and 3 is acceptable, because the testing should be conducted as close as possible to the conditions that would be experienced by a DG following an accident. Loading the DG solely with the inductive characteristics of a large motor will create a power factor less than unity. The design of the DG is set for full power operation with a power factor of ≥ 0.8. Therefore, testing of the DG for a loss of the single largest load and at full load is acceptable with a power factor ≤ 0.9. This change is designated as more restrictive because the testing required by the CTS does not currently contain these limitations.

M06 Unit CTS 4.8.1.1.2.e.4 and Unit 2 CTS 4.8.1.1.2.e.5 require each DG to auto-start on an ESF actuation signal and operate for greater than or equal to 5 minutes. ITS SR 3.8.1.11 requires the verification that each DG auto-start from standby condition and, in part, verify d) permanently connected loads remain energized from the offsite power system; and e) emergency loads are energized from the offsite power system. This changes the CTS by adding additional performance requirements for the ESF actuation test.

The purpose of the Unit 1 CTS 4.8.1.1.2.e.4 and Unit 2 CTS 4.8.1.1.2.e.5 is to test the performance of each DG on an ESF actuation signal. The proposed change adds explicit requirements to verify the permanently connected loads and

emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power. The change is acceptable because verifying the connection of the permanent loads and emergency loads will confirm the DG loading logic. This change is designated as more restrictive because additional acceptance criteria have been added to the CTS.

M07 CTS 4.8.1.1.2.f requires the diesel generators to be started simultaneously and accelerate to approximately 900 rpm in less than or equal to 10 seconds. ITS SR 3.8.1.18 requires the diesel generators to achieve voltage ≥ 3950 V and frequency ≥ 59.4 Hz and steady state voltage ≥ 3950 V and ≤ 4370 V and frequency ≥ 59.4 Hz and ≤ 60.6 Hz in ≤ 10 seconds. This changes the CTS by replacing the 900 rpm criterion with generator voltage and frequency requirements.

The purpose of CTS 4.8.1.1.2.f is to demonstrate that diesel generator starting independence has not been compromised and that each engine can achieve proper generator output within the specified time when the diesel generators are started simultaneously. The proposed changes add explicit requirements that verify the engine achieves design speed (i.e., proper frequency) and the generator output satisfies the design voltage and frequency requirements. The change is acceptable because verification that proper diesel generator voltage and frequency is achieved within 10 seconds confirms diesel generator starting independence has not been compromised. This change is designated as more restrictive because additional acceptance criteria is added to the CTS. Additional changes to CTS 4.8.1.1.2.f are discussed in DOC L11 and DOC L12.

M08 **Unit 2 only:** CTS 4.8.1.1.2.e.3 requires verification of generator capability to reject a load of 3685 kW without tripping and shall not exceed 4784 volts during and following the load rejection. The Surveillance does not specify that the testing be performed under certain unit conditions, nor that a DG shall be tested at a specific power factor. ITS SR 3.8.1.19 requires the verification that each DG does not trip, and voltage is maintained ≤ 4784 V during and following a load rejection of ≥ 3669 kW. Two Notes modify the SR. Note 1 states that the Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Finally, credit may be taken for unplanned events that satisfy this SR. Note 2 states that if the Surveillance is performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. This changes the CTS requirement by specifying conditions for when the DGs are normally tested and by specifying a power factor of ≤ 0.9, if the testing is conducted by synchronizing with the offsite sources.

The addition of Note 1 is acceptable, because performance of the Surveillance for the DG could cause perturbations to the electrical distribution systems. Restricting normal performance of the Surveillance on the DG during MODES 1 and 2 minimizes challenges to continued steady state operation and, as a result, the unit's safety systems. However, if an assessment of the risks indicates plant safety is maintained or enhanced, the Surveillance can be performed to

reestablish OPERABILITY. The addition of Note 2 is acceptable, because the testing should be conducted as close as possible to the conditions that would be experienced by a DG following an accident. Loading the DG solely with the inductive characteristics of a large motor will create a power factor less than unity. The design of the DG is set for full power operation with a power factor of ≥ 0.8. Therefore, testing of the DG for a loss of the single largest load and at full load is acceptable with a power factor ≤ 0.9. This change is designated as more restrictive because the testing required by the CTS does not currently contain these limitations.

M09 Unit 1 CTS 4.8.1.1.2.e.6 and Unit 2 CTS 4.8.1.1.2.e.7 require verification that each DG can operate for at least 24 hours within a specified load band. For the first two hours the Unit 1 load band is specified as 3800 kW to 3960 kW and 3800 kW to 3985 kW for Unit 2. ITS 3.8.1.13 specifies a two hour load band of 3800 kW to 3860 kW for Unit 1 DGs and 3800 kW to 3934 kW for the Unit 2 DGs. This changes the CTS by lowering the required upper load limit for the initial 2 hours of the DG 24 hour load run.

The purpose of Unit 1 CTS 4.8.1.1.2.e.6 and Unit 2 CTS 4.8.1.1.2.e.7 is to demonstrate that the DG can start and run continuously for not less than 24 hours at a full load capability with greater than or equal to 2 hours of the run at a load approximately equivalent to the 109% of the DG continuous rating for Unit 1 and approximately 104% of the DG continuous rating for Unit 2. The remainder of the run time is at a load equivalent to the continuous duty rating of the DG. The proposed change restricts the 2 hour band to a lower upper load band limit. The change is acceptable because the test loading is maintained consistent with the intent of RG 1.108 and within the 4 hour rating of the Unit 1 DGs and the 2000 hour rating of the Unit 2 DGs. This change is designated as more restrictive because it imposes a more restrictive load band on the DG test in lowering the upper band limit of the 2 hour load run. See DOC M11 for discussion of additional changes to the Unit 2 DG loading criteria.

M10 **Unit 2 only:** CTS 4.8.1.1.2.a.5, 4.8.1.1.2.e.7 and 4.8.1.1.2.e.13 specify a load band of 3450 kW to 3685 kW and CTS 4.8.1.1.2.e.3 specifies a generator load of 3685 kW for the full load rejection test. ITS SR 3.8.1.3, SR 3.8.1.13 and SR 3.8.1.20 specify bands of 3450 kW to 3669 kW, respectively. ITS SR 3.8.1.19 specifies a generator load of 3669 kW for the full load rejection test. This changes the CTS by changing the full load limit for testing of the Unit 2 DGs from 3685 kW to 3669 kW for the listed tests.

Analysis of the Unit 2 DGs identified the machines are engine limited (due to temperature limitations) at a continuous generator rating of 3669.4 kW and the DGs were derated. The change is acceptable because ITS retains testing of the DGs at the full load capability of the respective machine and is consistent with the RG 1.108 criteria. This change is designated more restrictive because it imposes a more restrictive load limit on the Unit 2 DGs.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.8.1.1.a requires two "physically independent" circuits between the offsite network and the onsite Class 1E AC Electrical Distribution System to be OPERABLE. CTS 3.8.1.1.b requires two "separate and independent" DG sets to be OPERABLE each with, in part, a separate fuel transfer pump. CTS 4.8.1.1.1 requires the determination of OPERABILITY for each "independent" circuit between the offsite transmission network and the onsite Class 1E AC Electrical Distribution System. In addition, a number of Surveillance Requirements in CTS have an **** footnote that states, “This test may be conducted in accordance with the manufacturer’s recommendations concerning engine prelube period.” ITS LCO 3.8.1 requires two qualified circuits between the offsite network and the onsite Class 1E AC Electrical Distribution System, two DGs capable of supplying the onsite Class 1E AC Electrical Distribution System. ITS SR 3.8.1.1 requires verification of the correct breaker alignment for each offsite circuit. The ITS SR Note regarding engine prelube for DG start Surveillances states, “All DG starts may be preceded by an engine prelube period.” This changes the CTS by moving the details that the offsite circuits are "independent" or "physically independent," that the DGs are "separate and independent" with a separate fuel transfer pump, and that engine prelube is conducted “in accordance with the manufacturer’s recommendations,” from the CTS to the ITS Bases.

The removal of these details related to system design from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for OPERABLE offsite sources and DGs and Surveillance Requirements to periodically verify the AC sources are OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.8.1.1 Action f describes specific Required Actions with one Unit 1 (Unit 2) startup transformer inoperable and with a Unit 2 (Unit 1) startup transformer connected to the same A or B offsite power circuit but administratively available to both units. ITS does not include specific Required Actions for a plant configuration utilizing the other unit’s startup transformer. This changes the CTS by moving the description of the plant configuration that is acceptable for a qualified offsite circuit using the opposite unit’s startup transformer from the CTS to the ITS Bases.

CTS 3.8.1.1 Action f was implemented as part of Amendment No. 103, “St. Lucie Unit 1 – Issuance of Amendment Re: Diesel Generator Reliability (TAC No.

75505)," dated June 7, 1990 (ADAMS Accession No. ML013550091) and Amendment No. 39, "St. Lucie Unit 2 – Issuance of Amendment Re: Diesel Generator Reliability (TAC Nos. 59634 and 64190)," dated February 7, 1989 (ADAMS Accession No. ML013600246). As stated in the safety evaluation supporting the amendments, the NRC determined the change was acceptable.

The described configuration relies on a two-unit facility design that allows for alignment of one unit's startup transformer to satisfy one of the two qualified offsite circuits requirement of CTS 3.8.1.1 for the other unit. The opposite unit startup transformer may be available to both units provided it is not supplying power to its respective unit Class 1E Electrical Distribution System. The design allows this alternate alignment as an acceptable qualified offsite circuit for the operating unit.

The removal of this detail related to system design from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. This change is acceptable because the ITS continues to require two qualified circuits between the offsite transmission network to the onsite Class 1E Electrical Distribution System to be OPERABLE and the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications. See DOC L01 for additional discussion of changes to CTS 3.8.1.1 Action f.

LA03 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Actions 3.8.1.1.b.2 and 3.8.1.1.c.2 include detail addressing the steam driven auxiliary feed pump as a required redundant feature. ITS Required Actions B.2 and C.2 do not include detail for specific redundant required features. This changes the CTS by moving the steam driven auxiliary feed pump detail from the CTS to the ITS Bases.

The removal of this detail related to system design from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA04 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Actions 3.8.1.1.b.3 and 3.8.1.1.c.3 require demonstration of the OPERABILITY of the OPERABLE DG by performance of Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours and clarifies the requirement by stating "If the EDG became inoperable due to any cause other than an inoperable

support system, an independently testable component, or preplanned preventative maintenance or testing." ITS Required Actions B.3.1 and B.3.2 do not include the clarifying detail. This changes the CTS by moving the clarifying detail from the CTS to the ITS Bases. | R1

The removal of this detail related to system design from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to the Bases to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA05 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 4.8.1.1.2.a.6 requires the verification that each DG is aligned to provide standby power to the associated emergency busses. ITS 3.8.1 does not contain this requirement. This changes the CTS by moving the detail that each DG is aligned to provide standby power to the associated emergency busses from the CTS to the ITS Bases.

The removal of these details that are related to system design from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for OPERABLE DGs. An OPERABLE DG must be capable of providing power to the associated emergency busses as indicated in the Bases. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA06 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* Unit 1 CTS 4.8.1.1.2.e.5.c and Unit 2 CTS 4.8.1.1.2.e.6.c require all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection signal. ITS SR 3.8.1.12 requires a similar verification, but instead of specifying the DG automatic trips that are not bypassed, the Surveillance refers to the trips that are bypassed as "noncritical." This changes the CTS by moving the detail of the specific DG trips that are not bypassed to the UFSAR. | R1

The removal of these details related to system design from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to verify that the

noncritical DG trips are bypassed on a loss of voltage signal concurrent with a safety injection signal. This change is acceptable because the UFSAR contains a complete listing of the DG electrical and mechanical trips and indicates the trips that are required whenever the DGs are required and the trips that are only required during the exercise mode of operation. In addition, PSL procedures periodically verify the capability of the DG trip relays and associated instrumentation to actuate a DG trip. The removed requirements will be adequately controlled in the UFSAR as any changes to the UFSAR are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA07 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 4.8.1.1.2.g requires the performance of pressure tests of those portions of the diesel fuel oil system designed to USAS B31.7 Class 3 requirements in accordance with the Inservice Inspection Program. Performance of the requirement is in accordance with the Surveillance Frequency Control Program. ITS 3.8.1 does not include this requirement. This changes the CTS by moving this requirement for the diesel fuel oil system to the Inservice Inspection Program.

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. Section XI of the ASME Boiler and Pressure Vessel Code contains the inservice inspection requirements addressed by this detail and the St. Lucie Inservice Inspection (ISI) Program implements the requirements. This change is acceptable because the removed information will be adequately controlled in the ISI Program. Any changes to the ISI Program are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information is being removed from the Technical Specifications.

LA08 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 4.8.1.1.4 requires that the Class 1E underground cable system be demonstrated OPERABLE within 30 days after the movement of any load in excess of 80% of the ground surface design basis load over the cable ducts with the method of test specifically described in the Surveillance Requirement. ITS 3.8.1 does not include this Surveillance Requirement. This changes the CTS by moving this requirement for the underground cable system to the UFSAR.

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. This change is acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.8.1.1 Actions b and c are modified by a NOTE requiring the completion of Surveillance Requirement 4.8.1.1.2.a.4 regardless of when the inoperable diesel generator is restored to operability if the absence of a common-cause failure cannot be confirmed. ITS 3.8.1 ACTION B requires a determination that the OPERABLE DG is not inoperable due to common cause failure through performance of SR 3.8.1.2 (CTS 4.8.1.1.2.a.4) unless it can be confirmed that the cause of the inoperable diesel generator does not exist on the OPERABLE diesel generator. However, ITS does not specify that the Required Action be completed if the Condition is exited. This changes the CTS by removing a requirement to complete the Surveillance Requirement once the ACTION is entered.

The purpose of CTS 3.8.1.1 Actions b and c is to provide compensatory measures to be taken in response to inoperable DG set(s), including demonstration that the OPERABLE DG is not inoperable due to a common cause. The Note requiring completion of the actions to verify the remaining DG is OPERABLE once the CTS Action is entered was added to the CTS with License Amendments 103 (Unit 1) and 39 (Unit 2) as part of the change to reflect the recommendations contained in Generic Letter 84-15. The requirement to complete the DG start test on the OPERABLE DG regardless of when the inoperable DG is restored to OPERABLE status was not included in ISTS 3.8.1 ACTION B. The requirement to complete the actions to demonstrate the OPERABLE DGs are not inoperable is not a required exception to ITS LCO 3.0.2 (CTS 3.0.2). Completing the ACTIONS to verify the OPERABILITY of the remaining OPERABLE DGs is not a required action to restore compliance with the LCO. Once a DG is restored to OPERABLE status, the need to start the OPERABLE DG if it cannot be determined the inoperable DG was not inoperable due to a common cause failure becomes unnecessary. Common cause failure evaluations are required by the PSL corrective action program pursuant to the requirements of 10 CFR 50, Appendix B. This change is acceptable because performance of a common cause failure evaluation is tracked in the plant's corrective action program, which if necessary, could require testing of the OPERABLE DG to confirm OPERABILITY. Additionally, deletion of the requirement to perform a start of the OPERABLE DG following restoration of the inoperable DG minimizes mechanical stress and wear on the OPERABLE DG engine and components. This change is designated as less restrictive, because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 (*Category 3 - Relaxation of Completion Time*) CTS 3.8.1.1 Action b.3 specifies the compensatory actions for one inoperable DG and CTS 3.8.1.1 Action c.3 specifies the compensatory actions for one inoperable offsite circuit and one inoperable DG. The Actions include a requirement to demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours. ITS 3.8.1 Required Action B.3.2 allows 24 hours to perform a similar check (ITS SR 3.8.1.2) on the remaining

OPERABLE DG. This changes the CTS by extending the time to perform this check from 8 hours to 24 hours.

The purpose of the CTS Action requirements to perform CTS 4.8.1.1.2.a.4 is to ensure that the other EDG is not inoperable because of a similar, yet undetected, failure (i.e., due to a common mode failure). The proposed 24 hour Completion time to start the OPERABLE DG is acceptable because it is considered a reasonable time to complete the DG start test on the other DG. Generic Letter 84-15 identified that a 24 hour time limit was acceptable to perform common mode failure checks or verify the DG start capability. In addition, the change is considered acceptable since the vast majority of DG start tests demonstrate that the DG is in fact OPERABLE. This change is designated as less restrictive because more time will be allowed to complete a Required Action in the ITS than is allowed in the CTS.

L03 *(Category 4 – Relaxation of Required Action)* CTS 3.8.1.1 Action f requires actions consistent with CTS 3.8.1.1 Action a except when the CTS 3.8.1.1.f actions are not met, the unit must be placed in COLD SHUTDOWN within the following 30 hours instead of HOT SHUTDOWN within the following 6 hours as required by CTS 3.8.1.1 Action a. ITS 3.8.1, ACTION G requires the unit to be placed in MODE 4 within 12 hours (i.e., HOT SHUTDOWN within 6 hours following HOT STANDBY in 6 hours) when the Required Actions and associated Completion Times of Condition A (i.e., one offsite circuit inoperable) are not met. This changes the CTS by changing the end state of CTS 3.8.1.1 Action f from "COLD SHUTDOWN within the following 30 hours" to "MODE 4 within 12 hours." (See DOC LA02 for discussion of removal of detail from CTS 3.8.1.1 Action f.)

The purpose of CTS 3.8.1.1 Actions a and f is to ensure that the remaining required AC source is OPERABLE when it is the only available offsite circuit to the affected unit and ensures that the inoperable startup transformer (offsite circuit) is restored to OPERABLE status within a reasonable time. CTS 3.8.1.1 Action f allows an opposite unit startup transformer to be used as an alternative AC source to support the qualified offsite circuit when the unit startup transformer is inoperable provided the opposite unit startup transformer is not supplying power to its own loads. If the opposite unit startup transformer is subsequently needed to supply its own loads, CTS 3.8.1.1 Action f duplicates Required Actions provided in CTS 3.8.1.1.a.1 and 3.8.1.1.a.3 and contrary to the Required Actions for one offsite circuit inoperable provided in Action a requires the unit to be in COLD SHUTDOWN within 36 hours if the Required Actions of CTS 3.8.1.1.f cannot be satisfied. This change is acceptable because the impact of not restoring the offsite circuit to Operable status is not changed based on the AC source (unit startup transformer or opposite unit startup transformer). The change to the end state from MODE 5 to MODE 4 is consistent with the Required Actions of CTS 3.8.1.1 Action a as revised in Amendments 234 and 184 (Units 1 and 2, respectively), "St. Lucie Plant, Unit Nos. 1 and 2 – Issuance of Amendments Regarding Changes in Selected Technical Specifications End States (CAC Nos. MF8106 and MF8107; Formerly CAC NOS. MF6683 AND MF6684)," dated August 30, 2016, (ADAMS Accession No. ML16210A374). In addition, the change of the end state from MODE 5 to MODE 4 is acceptable because the modification is consistent with CE-NPSD-1186-A, Revision 00, "Technical Justification for the Risk-Informed Modification to Selected Required

Action End States for CEOG Member PWRs," dated October 2001 (ADAMS Accession No. 110410539). This change is designated as less restrictive because instead of requiring the plant to achieve MODE 5 (COLD SHUTDOWN), the ITS end state is MODE 4 (HOT SHUTDOWN) with LCO 3.0.4.a allowance not applicable.

L04 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.8.1.1.2.a.4 and CTS 4.8.1.1.2.f require verification that the DG starts from ambient condition and accelerates to approximately 900 rpm in less than or equal to 10 seconds. ITS SR 3.8.1.2, SR 3.8.1.7 and SR 3.8.1.18 require similar tests but do not include the specific criteria to accelerate to 900 rpm. In addition, the voltage and frequency criteria during the 10 second start is modified in CTS 4.8.1.1.2.a.4, CTS 4.8.1.1.2.e.4.a (Unit 1), and CTS 4.8.1.1.2.e.5.a (Unit 2). ITS SR 3.8.1.2 (DG slow start test) requires that each DG starts from standby conditions and achieves a steady state voltage of ≥ 3950 V and ≤ 4370 V and ≥ 59.4 Hz and ≤ 60.6 Hz. ITS SR 3.8.1.7 (DG quick start test), ITS SR 3.8.1.18, and ITS SR 3.8.1.20 (Unit 2), in part, requires the verification that each DG starts from standby conditions and achieves a voltage of ≥ 3950 V and frequency ≥ 59.4 Hz within 10 seconds and achieves a steady state voltage of ≥ 3950 V and ≤ 4370 V and ≥ 59.4 Hz and ≤ 60.6 Hz. ITS SR 3.8.1.11 (DG ESF actuation test), in part, requires the verification that each DG starts from standby conditions and achieves a voltage of ≥ 3950 V and frequency ≥ 59.4 Hz within 10 seconds and achieves a steady state voltage of ≥ 3950 V and ≤ 4370 V and ≥ 59.4 Hz and ≤ 60.6 Hz. This changes CTS 4.8.1.1.2.a.4 by removing the 900 rpm and the 10 seconds to achieve the steady state voltage and frequency range as addressed in ITS SR 3.8.1.2 criterion. It changes CTS 4.8.1.1.2.a.4, CTS 4.8.1.1.2.e.4.a (Unit 1), and CTS 4.8.1.1.2.e.5.a (Unit 2) by specifying a minimum voltage and frequency to be achieved within 10 seconds instead of requiring the voltage and frequency to be within range in 10 seconds. This effectively allows the upper voltage and frequency limits to be exceeded during DG acceleration and stabilization.

The purpose of the CTS 4.8.1.1.2.a.4, CTS 4.8.1.1.2.e.4.a (Unit 1), and CTS 4.8.1.1.2.e.5.a (Unit 2) is to test for the OPERABILITY of each DG to start and load on an ESF signal, respectively. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria (i.e., upper voltage and frequency limits) are not necessary for verification that the equipment used to meet the LCO can perform its required functions. This change includes narrowing the current voltage and frequency ranges consistent with the DG steady state voltage and frequency ranges specified in other CTS Surveillances (e.g., CTS 4.8.1.1.2.e.3.b (Unit 1) and CTS 4.8.1.1.2.e.4.b (Unit 2)). This changes the CTS by specifying a minimum voltage and frequency limit to be achieved within 10 seconds instead of a voltage and frequency range. This effectively allows the upper voltage and frequency limits to be exceeded during DG acceleration and stabilization. As stated herein, proposed SR 3.8.1.7, SR 3.8.11 and SR 3.8.18 will require only the establishment of the minimum frequency and voltage within the given time frame. The accident analyses and other Surveillances require that the DGs be capable of being loaded within 10 seconds. This can be accomplished at the lowest level in the frequency and voltage ranges. While the upper level requirement regarding the frequency and voltage acceptance criterion is being eliminated during the 10 second start, the

requirement to establish a steady state voltage and frequency has been retained. Once steady state conditions are reached, the minimum and maximum voltage and frequency limits must be maintained. Verification that the minimum voltage and frequency limits are met within the proper time is sufficient to ensure the DG can perform its design function. When called upon, the DG must start and accept load within the proper time. Once the minimum voltage and frequency limits are met, the DG can connect to the bus. When a test is performed that does not result in connecting the DG to the bus, a voltage or frequency overshoot can occur since no loads are connected (the loading tends to minimize overshoot). This overshoot could be such that the voltage or frequency is outside the band high when the time limit expires. However, this condition is not indicative of an inoperable DG, provided that steady state voltage and frequency are maintained. The DG start times are monitored and trend evaluated to identify degradation of DG governor and voltage regulator performance as described in the ITS Bases. This change is designated as less restrictive because less stringent Surveillance test criteria are being applied in the ITS than were applied in the CTS.

L05 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.8.1.1.2.a.4 requires verification that each DG starts and achieves voltage and frequency within the specified limits. Additionally, the Surveillance requires the DG to be started for the test by using one of the following signals with startup on each signal verified in accordance with the Surveillance Frequency Control Program: manual/local, simulated loss-of-offsite power by itself, simulated loss-of-offsite power in conjunction with an ESF actuation signal, and an ESF actuation signal by itself. ITS SR 3.8.1.2 requires a similar test but does not include the requirement to vary the start signal between manual, loss of offsite power, simulated loss-of-offsite power in conjunction with an ESF actuation signal, and ESF actuation. This changes the CTS by deleting the requirement to use different start signals in the performance of the DG start Surveillance.

The purpose of CTS 4.8.1.1.2.a.4 is to verify the capability of each DG to start and achieve voltage and frequency with the specified limits. This change is acceptable because the deleted requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between the origin of the start signal. Therefore, the results of the test are unaffected by the type of signal used to initiate the test. Each DG will continue to be tested in a manner to ensure the safety analyses assumption will be met. This change is designated as less restrictive because a Surveillance which is required in the CTS will not be required in the ITS.

L06 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.8.1.1.2.a.5 requires verification that each DG is synchronized, loaded to greater than or equal to 3500 kW in accordance with manufacturer's recommendations and operates within a load band of 3300 to 3500 kW for at least an additional 60 minutes. Unit 2 CTS 4.8.1.1.2.a.5 requires verification that each DG is synchronized, loaded to greater than or equal to 3685 kW in accordance with manufacturer's recommendations and operates within a load band of 3450 to 3685 kW for at least an additional 60 minutes. ITS SR 3.8.1.3 requires a similar test but does not include the requirement to initially load the diesel generator to greater than or equal to 3500 kW (Unit 1) or 3685 kW (Unit 2). This changes the CTS by deleting the requirement to initially load the diesel

generator to greater than or equal to 3500 kW (Unit 1) and 3685 kW (Unit 2). See DOC M11 for additional discussion of changes.

The purpose of CTS 4.8.1.1.2.a.5 (Unit 1 and Unit 2) is to verify the capability of each DG to synchronize to the offsite electrical system and accept loads greater than or equal to the equivalent of the maximum expected accident loads with a minimum run time to stabilize temperatures. This change is acceptable because the relaxed Surveillance Requirement acceptance criterion is not necessary to verify that the equipment used to meet the LCO can perform its required functions. This change is designated as less restrictive because a Surveillance test criterion, which is required in the CTS, will not be required in the ITS.

L07 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.8.1.1.2.b.1 requires the removal of accumulated water from the engine-mounted fuel tanks at least in accordance with the Surveillance Frequency Control Program and "after each occasion when the diesel is operated for greater than 1 hour." ITS SR 3.8.1.5 requires the same Surveillance but does not include requirement of after each occasion when the diesel is operated for greater than 1 hour. This changes the CTS by deleting the conditional requirement to test for accumulated water after each operation of the DG for greater than one hour.

The purpose of CTS 4.8.1.1.2.b.1 is to provide a degree of assurance that the engine-mounted fuel tanks are free of accumulated water each time the associated DG is operated for an hour or more.

This change is acceptable because the requirement to check for and remove accumulated water is retained in the ITS. Water condensation within the engine-mounted fuel tanks is a time dependent process, not a process dependent on the transfer of fuel oil during DG operation. Since it is the expectation that the DG will not be operated except for the nominal monthly OPERABILITY tests (based on experience), and that the fuel oil storage tanks are also periodically checked for water, no increased Frequency is necessary. This change is designated as less restrictive because the explicit requirement to remove accumulated water in the engine-mounted fuel tanks after a DG run of greater than one hour has been deleted.

L08 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* CTS 4.8.1.1.2.e.2 requires verification of the generator capability to reject a load greater than or equal to the specified value while maintaining voltage within ± 10 percent of the initial pretest voltage and frequency at 60 ± 1.2 Hz. ITS SR 3.8.1.9 requires verification that following load rejection, frequency is ≤ 66.7 Hz, and within 1.8 seconds following load rejection voltage is maintained ≥ 3950 V and ≤ 4370 V and frequency is maintained ≥ 59.4 Hz and ≤ 60.6 Hz. This changes the CTS in that the maximum frequency is relaxed to 66.7 Hz following load rejection and the time to reach the DG steady state frequency and voltage range is added. The voltage and frequency ranges are consistent with the DG steady state voltage and frequency ranges specified in other CTS Surveillances (e.g., CTS 4.8.1.1.2.e.3.b (Unit 1) and CTS 4.8.1.1.2.e.4.b (Unit 2)).

The purpose of CTS 4.8.1.1.2.e.2 is to ensure the proper operation of the DG governor and load control circuits. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria is not necessary for verification that the equipment used to meet the LCO can perform its required functions. These changes the CTS allow the transient frequency following load rejection and time to recover the voltage and frequency. Consistent with Regulatory Guide 1.9, the load rejection test is acceptable if the diesel speed does not exceed 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal speed, whichever is lower. The overspeed trip of the PSL DGs is 1035 rpm, which is 115% of the DG synchronous speed. A frequency of 66.7 Hz equates to the synchronous speed plus 75% of the difference between synchronous speed and the overspeed trip setpoint (i.e., 1001.25 rpm). Consistent with Regulatory Guide 1.9, the time allowed to recover the voltage and frequency to within the nominal range is calculated based on 60% of the load sequence interval of the largest single load. The proposed timing of 1.8 seconds equates to 60% of the load sequence interval of the single largest load; 3 seconds. This change is acceptable since the DG continues to be tested consistent with the recommendations of Regulatory Guide 1.9. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L09 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.8.1.1.2.e.5 and Unit 2 CTS 4.8.1.1.2.e.6 require verification of DG performance following an actual or simulated loss of offsite power in conjunction with an ESF actuation signal. ITS SR 3.8.1.12 and ITS SR 3.8.1.17 specify that the signal may be from either "actual or simulated" actuation signals. This changes the CTS by explicitly allowing the use of either an actual or simulated signal for the actuation signals and changes the specific actuation signal (i.e., safety injection) to ESF.

The purpose of Unit 1 CTS 4.8.1.1.2.e.5 and Unit 2 CTS 4.8.1.1.2.e.6 is to ensure that the AC Sources operate correctly upon receipt of an actuation signal. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an actual or simulated signal therefore the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. In addition, ITS retains the specific ESF actuation signal (i.e., safety injection signal) information in the ITS Bases. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L10 *(Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.8.1.1.2.e.6 and Unit 2 CTS 4.8.1.1.2.e.7 requires verification that each DG can operate for at least 24 hours, with loading within the specified load bands.

The CTS further requires that voltage and frequency be 4160 ± 240 volts and 60 ± 1.2 Hz within 10 seconds after the start signal and steady state generator voltage and frequency shall be maintained within these limits during this test. ITS SR 3.8.1.13 requires similar tests however it does not specify voltage and frequency limits for the test. This changes the CTS by removing the voltage and frequency limits from the Surveillance.

The purpose of Unit 1 CTS 4.8.1.1.2.e.6 and Unit 2 CTS 4.8.1.1.2.e.7 is to verify the capability of each DG to start and run continuously at full load for at least 24 hours. DG operation with voltage and frequency within the specified limits will continue to be verified during performance of Unit 1 ITS SR 3.8.1.2, SR 3.8.1.7, SR 3.8.1.9, SR 3.8.1.10, SR 3.8.1.11, SR 3.8.1.17, and SR 3.8.1.18; and Unit 2 ITS SR 3.8.1.2, SR 3.8.1.7, SR 3.8.1.9, SR 3.8.1.10, SR 3.8.1.11, SR 3.8.1.17, SR 3.8.1.18 and SR 3.8.1.20. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. The Surveillance will continue to verify that the DG is capable of running continuously at full load for an interval of not less than 24 hours. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L11 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.8.1.1.2.f requires verification that the DGs accelerate to 900 rpm within 10 seconds when started simultaneously in accordance with the Surveillance Frequency Control Program or after any modification which could affect DG interdependence. ITS SR 3.8.1.18 does not include the requirement to test simultaneous start of the DGs after any modifications which could affect DG interdependence. This changes the CTS by deleting the requirement to simultaneously start the DGs after any modifications which could affect DG interdependence.

This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO is consistent with the safety analyses. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the assumptions in the safety analyses are protected. Following repair, maintenance, modification, or replacement of a component that may affect OPERABILITY, post maintenance testing is required to demonstrate OPERABILITY of the system or component. This is described in the Bases of ITS SR 3.0.1 and required under SR 3.0.1. The OPERABILITY requirements of the DGs are described in the Bases for Specification 3.8.1. In addition, the requirements of 10 CFR 50, Appendix B, Section XI (Test Control) provide adequate controls for test programs to ensure that testing incorporates applicable acceptance criteria. Compliance with 10 CFR 50 is required under the unit's operating license. As a result, post maintenance testing will continue to be performed and an explicit requirement in the Technical Specifications is not necessary. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS. Additional changes to CTS 4.8.1.1.2.f are discussed in DOC M07, DOC L04 and DOC L12.

L12 *(Category 8 – Deletion of Surveillance Requirement Shutdown Performance Requirements)* CTS 4.8.1.1.2.f contains the requirement to perform the test "during shutdown." ITS SR 3.8.1.18 does not specify conduct of the test to occur during shutdown. This changes the CTS by deleting the requirement to perform the Surveillance during shutdown.

The purpose of CTS 4.8.1.1.2.f is to demonstrate that diesel generator starting independence has not been compromised and that each engine can achieve proper speed within the specified time when the diesel generators are started simultaneously. The proposed Surveillance does not include the restriction that it be performed during shutdown. The control of unit conditions appropriate to perform the test is an issue for procedures and scheduling and has been determined by the NRC Staff to be unnecessary as a Technical Specification restriction. As indicated in Generic Letter 91-04, removal of this specific restriction is consistent with the vast majority of other Technical Specification Surveillances that do not dictate unit conditions for the Surveillance. This change is designated as less restrictive because the Surveillance may be performed during plant conditions other than shutdown. Additional changes to CTS 4.8.1.1.2.f are discussed in DOC M07, DOC L04 and DOC L11.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

3.8.1.1 LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

3.8.1.1.a a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System~~,~~ and

3.8.1.1.b b. Two diesel generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System~~, and~~

~~[c. Automatic load sequencers for Train A and Train B.]~~

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

----------NOTE----------

Action g LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.1 A. One ~~[required]~~ offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for ~~[required]~~ OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND	
Action a.2	A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND	

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ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.3		A.3 Restore ~~[required]~~ offsite circuit to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
Action b.1	B. One ~~[required]~~ DG inoperable.	B.1 Perform SR 3.8.1.1 for the OPERABLE ~~[required]~~ offsite circuit(s).	1 hour AND Once per 8 hours thereafter
		AND	
Action b.2		B.2 Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
		AND	
Action b.3 DOC L02		B.3.1 Determine OPERABLE DG(s) is not inoperable due to common cause failure.	~~[~~24~~]~~ hours
		OR	
Action b.3 DOC L02		B.3.2 Perform SR 3.8.1.2 for OPERABLE DG(s).	~~[~~24~~]~~ hours
		AND	

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.4 Restore ~~[required]~~ DG to OPERABLE status.	14 days ~~72 hours~~ ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
C. Two ~~[required]~~ offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. AND C.2 Restore one ~~[required]~~ offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) 24 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
D. One ~~[required]~~ offsite circuit inoperable. AND One ~~[required]~~ DG inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," when Condition D is entered with no AC power source to any train. ----------	

Action b.4
Action d.1
Action d.2
Action c
Action c NOTE 1

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3
3
3
3

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action c.4	D.1 Restore [required] offsite circuits to OPERABLE status.	12 hours [OR In accordance with the Risk Informed Completion Time Program]
	OR	
Action c.4	D.2 Restore [required] DG to OPERABLE status.	12 hours [OR In accordance with the Risk Informed Completion Time Program]
Action e E. Two [required] DGs inoperable.	E.1 Restore one [required] DG to OPERABLE status.	2 hours
Insert 1 F. ~~NOTE This Condition may be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads following a loss of offsite power independent of, or coincident with, a Design Basis Event.~~ ~~[One [required] [automatic load sequencer] inoperable.~~	~~F.1 Restore [required] [automatic load sequencer] to OPERABLE status.~~	~~[12] hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~

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Insert 1 (5)

Action d.2

F. Required Action and associated Completion Time of Condition C not met.	F.1 Be in MODE 3.	6 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.3 Action b.4 Action c.4 Action e Action f G. Required Action and associated Completion Time of Condition A, B, ~~C,~~ D, ~~E,~~ or ~~[F]~~ not met. (E)	G.1 Be in MODE 3. AND G.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours (5) (3) 12 hours
DOC M01 H. Three or more ~~[required]~~ AC sources inoperable.	H. 1 Enter LCO 3.0.3.	Immediately (3)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.1.1.1.a SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each ~~[required]~~ offsite circuit.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
4.8.1.1.2.a.4 Footnote ** SR 3.8.1.2	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 2. [A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met.] ----------	
4.8.1.1.2.a.4 DOC L04	Verify each DG starts from standby conditions and achieves steady state voltage ≥ [3740] V and ≤ [4580] V, and frequency ≥ [58.8] Hz and ≤ [61.2] Hz. (3950; 4370; 59.4; 60.6)	[31 days OR In accordance with the Surveillance Frequency Control Program] (3)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	----------NOTES---------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.7. ----------	
Verify each DG is synchronized and loaded, and operates for ≥ 60 minutes at a load ≥ ~~[4500]~~ 3300 kW and ≤ ~~[5000]~~ 3500 kW.		~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)
SR 3.8.1.4	Verify each day tank ~~[and engine mounted tank]~~ contains ≥ ~~[220]~~ 152 gal of fuel oil.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

4.8.1.1.2.a.5 Footnote **
DOC A04
4.8.1.1.2.a.5 Footnote ***
DOC M03
DOC M04
4.8.1.1.2.a.5
DOC L06
3.8.1.1.b.1
4.8.1.1.2.a.1

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.8.1.1.2.b.1 DOC L07	SR 3.8.1.5	Check for and remove accumulated water from each day tank ~~[and engine mounted tank]~~.	~~[[31] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3
4.8.1.1.2.a.3 DOC M02	SR 3.8.1.6	Verify the fuel oil transfer system operates to ~~[~~automatically~~]~~ transfer fuel oil from storage tank~~[~~s~~]~~ to the day tank ~~[and engine mounted tank]~~. (s)	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1 3
4.8.1.1.2.a.4 Footnote **	SR 3.8.1.7	----------NOTE---------- All DG starts may be preceded by an engine prelube period. ----------		
4.8.1.1.2.a.4 DOC L04		Verify each DG starts from standby condition and achieves: a. In ≤ ~~[~~10~~]~~ seconds, voltage ≥ ~~[3740]~~ V (3950) and frequency ≥ ~~[58.8]~~ Hz (59.4) and b. Steady state voltage ≥ ~~[3740]~~ V (3950) and ≤ ~~[4580]~~ V (4370), and frequency ≥ ~~[58.8]~~ Hz (59.4) and ≤ ~~[61.2]~~ Hz (60.6).	~~[184 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	~~NOTE~~ ~~[This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.]~~ 4.8.1.1.1.b ~~[~~Verify ~~[~~automatic ~~[~~and~~]~~ manual~~]~~ transfer of AC power sources from the normal ~~offsite~~ circuit to ~~each alternate [required]~~ offsite circuit. (the)	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~ (3) (1)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	----------NOTES---------- 1. ~~[~~This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 2. If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ ~~[~~0.9~~]~~. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable.~~]~~ ----------	
	Verify each DG rejects a load greater than or equal to its associated single largest post-accident load and:	~~[[18] months~~ ~~OR~~
	a. Following load rejection, the frequency is ≤ ~~[63]~~ Hz, b. Within ~~[3]~~ seconds following load rejection, the voltage is ≥ ~~[3740]~~ V and ≤ ~~[4580]~~ V, and c. Within ~~[3]~~ seconds following load rejection, the frequency is ≥ ~~[58.8]~~ Hz and ≤ ~~[61.2]~~ Hz.	In accordance with the Surveillance Frequency Control Program~~]~~

DOC M05

4.8.1.1.2.e.2 NOTE

DOC M05

4.8.1.1.2.e.2

DOC L08

66.7

1.8

1.8

3950

4370

59.4

60.6

3

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.10 ----------NOTES---------- 1. [This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 2. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ [0.9]. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable.] ---------- Verify each DG does not trip, and voltage is maintained ≤ [5000] V during and following a load rejection of ≥ [4500] kW and ≤ [5000] kW.	[[18] months OR In accordance with the Surveillance Frequency Control Program]

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CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.1~~1~~ 0	6
----------NOTES----------	
1. All DG starts may be preceded by an engine prelube period. (4.8.1.1.2.e.3.b footnote ****)	
2. This Surveillance shall not normally be performed in MODE 1, 2~~, 3, or 4~~. (or) However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. (4.8.1.1.2.e.3 NOTE)	7

Verify on an actual or simulated loss of offsite power signal: (4.8.1.1.2.e.3)	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~
a. De-energization of emergency buses, (4.8.1.1.2.e.3.a)	
b. Load shedding from emergency buses, (4.8.1.1.2.e.3.a)	
c. DG auto-starts from standby condition and: (4.8.1.1.2.e.3.b)	
1. Energizes permanently connected loads in ≤ ~~[~~10~~]~~ seconds,	
2. Energizes auto-connected ~~shutdown~~ (emergency) loads through ~~[automatic~~ load sequence~~r]~~, (timers)	3 1
3. Maintains steady state voltage ≥ ~~[3740]~~ (3950) V and ≤ ~~[4580]~~ (4370) V,	
4. Maintains steady state frequency ≥ ~~[58.8]~~ (59.4) Hz and ≤ ~~[61.2]~~ (60.6) Hz, and	
5. Supplies permanently connected ~~[~~and auto-connected~~]~~ ~~shutdown~~ (emergency) loads for ≥ 5 minutes.	1

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.1~~2~~ (1)	
----------NOTES----------	
1. ~~[~~ All DG starts may be preceded by an engine prelube period.	
2. This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.	

Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal each DG auto-starts from standby condition and:	~~[[18] months~~ ~~OR~~
a. In ≤ ~~[~~10~~]~~ seconds after auto-start and during tests, achieves voltage ≥ ~~[3740]~~ V and frequency ≥ ~~[58.8]~~ Hz, (3950; 59.4)	In accordance with the Surveillance Frequency Control Program~~]]~~
b. Achieves steady state voltage ≥ ~~[3740]~~ V and ≤ ~~[4580]~~ V and frequency ≥ ~~[58.8]~~ Hz and ≤ ~~[61.2]~~ Hz, (3950; 59.4; 4370; 60.6)	
c. Operates for ≥ 5 minutes,	
d. Permanently connected loads remain energized from the offsite power system, and	
e. Emergency loads are energized ~~[~~or auto-connected through the ~~automatic~~ load sequence~~r]~~ from the offsite power system. (timers)	

4.8.1.1.2.e.4 Footnote ****
4.8.1.1.2.e.4 NOTE
4.8.1.1.2.e.4 DOC A07
4.8.1.1.2.e.4.a DOC L04
4.8.1.1.2.e.4.c
4.8.1.1.2.e.4.b
DOC M06

(6) (3) (1)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.1~~3~~ (2)	----------NOTE---------- ~~[~~This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.~~]~~ ----------	(6)
4.8.1.1.2.e.5.c DOC L09	Verify each DG's noncritical automatic trips are bypassed on ~~[~~actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal~~]~~.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

4.8.1.1.2.e.5 NOTE

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.14 (3)	(6)
----------NOTES----------	
1. Momentary transients outside the load and power factor ranges do not invalidate this test.	
2. ~~This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced.~~ Credit may be taken for unplanned events that satisfy this SR.	R1 (1)
3. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ ~~[~~0.9~~]~~. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable.	(3)

Verify each DG operates for ≥ 24 hours:	~~[[18] months~~
	~~OR~~
a. For ≥ ~~[~~2~~]~~ hours loaded ≥ ~~[5250]~~ kW (3800) and ≤ ~~[5500]~~ kW (3860) and	In accordance with the Surveillance Frequency Control Program~~]~~ (3)
b. For the remaining hours of the test loaded ≥ ~~[4500]~~ kW (3300) and ≤ ~~[5000]~~ kW (3500).	

4.8.1.1.2.e.6 Footnote #

4.8.1.1.2.e.6 NOTE

DOC M05

4.8.1.1.2.e.6

DOC M09

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
~~SR 3.8.1.15~~ ~~----------NOTES----------~~ ~~1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ [2] hours loaded ≥ [4500] kW and ≤ [5000] kW.~~ ~~Momentary transients outside of load range do not invalidate this test.~~ ~~2. All DG starts may be preceded by an engine prelube period.~~ ~~----------~~ ~~Verify each DG starts and achieves:~~ ~~a. In ≤ [10] seconds, voltage ≥ [3740] V and frequency ≥ [58.8] Hz and~~ ~~b. Steady state voltage ≥ [3740] V and ≤ [4580] V, and frequency ≥ [58.8] Hz and ≤ [61.2] Hz.~~	~~[[18] months~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~
SR 3.8.1.1~~6~~ (4) ----------NOTE---------- This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon ~~a~~ simulated (an actual or) restoration of offsite power, b. Transfers loads to offsite power source, and c. Returns to ready-to-load operation.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

4.8.1.1.2.e.8 NOTE

4.8.1.1.2.e.8

4.8.1.1.2.e.8.a

4.8.1.1.2.e.8.b

4.8.1.1.2.e.8.c

(8) (8) (1) (3)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.17 (5) ----------NOTE---------- 4.8.1.1.2.e.9 NOTE [This Surveillance shall not normally be performed in MODE 1 (or) 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- 4.8.1.1.2.e.9 Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to ready-to-load operation and [b. Automatically energizing the emergency load from offsite power.]	[[18] months OR In accordance with the Surveillance Frequency Control Program]] (8, 7, 3)
SR 3.8.1.18 (6) ----------NOTE---------- 4.8.1.1.2.e.11 NOTE [This Surveillance shall not normally be performed in MODE 1 (or) 2, 3, or 4. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.] ---------- 4.8.1.1.2.e.11 Verify interval between each sequenced load block is within ± [10% of design interval] for each [emergency [and shutdown]] load sequencer. (1 second; timer)	[[18] months OR In accordance with the Surveillance Frequency Control Program] (8, 7, 3, 2, 1)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.1~~9~~ (7)	(8)
----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall (or) not normally be performed in MODE 1~~,~~ 2~~, 3, or 4~~. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. (7) ----------	
Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal:	~~[~~[18] months ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)
a. De-energization of emergency buses,	
b. Load shedding from emergency buses,	
c. DG auto-starts from standby condition and:	
1. energizes permanently connected loads in ≤ ~~[~~10~~]~~ seconds,	
2. energizes auto-connected emergency loads through ~~[~~load sequence~~r]~~, (timers)	(2)
3. achieves steady state voltage ≥ ~~[3740]~~ V and ≤ ~~[4580]~~ V, (3950) (4370)	
4. achieves steady state frequency ≥ ~~[58.8]~~ Hz and ≤ ~~[61.2]~~ Hz, and (59.4) (60.6)	
5. supplies permanently connected ~~[~~and auto-connected~~]~~ emergency loads for ≥ ~~[~~5~~]~~ minutes.	

4.8.1.1.2.e.5 Footnote ****

4.8.1.1.2.e.5 NOTE

4.8.1.1.2.e.5 DOC L09

4.8.1.1.2.e.5.a

4.8.1.1.2.e.5.a

4.8.1.1.2.e.5.b

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.~~20~~ (18) ----------NOTE---------- All DG starts may be preceded by an engine prelube period. ----------	(8)
Verify, when started simultaneously from standby condition, each DG achieves: a. In ≤ ~~[~~10~~]~~ seconds, voltage ≥ ~~[3740]~~ (3950) V and frequency ≥ ~~[58.8]~~ (59.4) Hz and b. Steady state voltage ≥ ~~[3740]~~ (3950) V and ≤ ~~[4580]~~ (4370) V, and frequency ≥ ~~[58.8]~~ (59.4) Hz and ≤ ~~[61.2]~~ (60.6) Hz.	~~[10 years~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

4.8.1.1.2.f Footnote ****

4.8.1.1.2.f

DOC M07
DOC L04
DOC L11
DOC L12

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

3.8.1.1 LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

3.8.1.1.a a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System~~,~~ and

3.8.1.1.b b. Two diesel generators (DGs) each capable of supplying one train of the onsite Class 1E AC Electrical Power Distribution System~~, and~~ (1)

~~[c. Automatic load sequencers for Train A and Train B.]~~ (3)

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

----------NOTE----------

Action g LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.1 A. One ~~[required]~~ offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for ~~[required]~~ OPERABLE offsite circuit.	1 hour (3) AND Once per 8 hours thereafter
	AND	
Action a.2	A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND	

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.3	A.3 Restore ~~[required]~~ offsite circuit to OPERABLE status.	72 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
B. One ~~[required]~~ DG inoperable.	B.1 Perform SR 3.8.1.1 for the OPERABLE ~~[required]~~ offsite circuit(s).	1 hour AND Once per 8 hours thereafter
Action b.1	AND	
Action b.2	B.2 Declare required feature(s) supported by the inoperable DG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND	
Action b.3 DOC L02	B.3.1 Determine OPERABLE DG(s) is not inoperable due to common cause failure.	~~[~~24~~]~~ hours
	OR	
Action b.3 DOC L02	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG(s).	~~[~~24~~]~~ hours
	AND	

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b.4	B.4 Restore ~~[required]~~ DG to OPERABLE status.	14 days ~~72 hours~~ ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~ (4, 3)
Action d.1 C. Two ~~[required]~~ offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. AND	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s) (3)
Action d.2	C.2 Restore one ~~[required]~~ offsite circuit to OPERABLE status.	24 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~ (3)
Action c D. One ~~[required]~~ offsite circuit inoperable. AND One ~~[required]~~ DG inoperable.	----------NOTE---------- Action c NOTE 1 Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," when Condition D is entered with no AC power source to any train. ----------	(3)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action c.4	D.1 Restore ~~[required]~~ offsite circuit~~s~~ to OPERABLE status. OR D.2 Restore ~~[required]~~ DG to OPERABLE status.	12 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~
Action e E. Two ~~[required]~~ DGs inoperable.	E.1 Restore one ~~[required]~~ DG to OPERABLE status.	2 hours
Insert 1 ~~F. ----------NOTE---------- This Condition may be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads following a loss of offsite power independent of, or coincident with, a Design Basis Event. ---------- [One [required] [automatic load sequencer] inoperable.~~	~~F.1 Restore [required] [automatic load sequencer] to OPERABLE status.~~	~~[12] hours [OR In accordance with the Risk Informed Completion Time Program]]~~

CTS

5

Insert 1

Action d.2

F. Required Action and associated Completion Time of Condition C not met.	F.1 Be in MODE 3.	6 hours

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a.3 Action b.4 Action c.4 Action e Action f G. Required Action and associated Completion Time of Condition A, B, ~~C,~~ D, ~~E,~~ or ~~[F]~~ not met. (E)	G.1 Be in MODE 3. AND G.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours (5) (3) 12 hours
DOC M01 H. Three or more ~~[required]~~ AC sources inoperable.	H. 1 Enter LCO 3.0.3.	Immediately (3)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.1.1.1.a SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each ~~[required]~~ offsite circuit.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
4.8.1.1.2.a.4 Footnote ** SR 3.8.1.2	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 2. ~~[~~A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met.~~]~~ ----------	
4.8.1.1.2.a.4 DOC L04	Verify each DG starts from standby conditions and achieves steady state voltage ≥ ~~[3740]~~ 3950 V and ≤ ~~[4580]~~ 4370 V, and frequency ≥ ~~[58.8]~~ 59.4 Hz and ≤ ~~[61.2]~~ 60.6 Hz.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
4.8.1.1.2.a.5 DOC A04	SR 3.8.1.3 ----------NOTES---------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 4.8.1.1.2.a.5 Footnote *** 2. Momentary transients outside the load range do not invalidate this test. DOC M03 3. This Surveillance shall be conducted on only one DG at a time. DOC M04 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.7. ----------	
4.8.1.1.2.a.5 DOC L06 DOC M10	Verify each DG is synchronized and loaded, and operates for ≥ 60 minutes at a load ≥ ~~[4500]~~ kW and ≤ ~~[5000]~~ kW. (3669) (3450)	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)
3.8.1.1.b.1 4.8.1.1.2.a.1	SR 3.8.1.4 Verify each day tank ~~[and engine mounted tank]~~ contains ≥ ~~[220]~~ gal of fuel oil. (238)	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.8.1.1.2.b.1 DOC L07	SR 3.8.1.5	Check for and remove accumulated water from each day tank ~~[and engine mounted tank]~~.	~~[[31] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3
4.8.1.1.2.a.3 DOC M02	SR 3.8.1.6	Verify the fuel oil transfer system operates to ~~[~~automatically~~]~~ transfer fuel oil from storage tank~~[~~s~~]~~ to the day tank ~~[and engine mounted tank]~~. (s)	~~[[92] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1 3
4.8.1.1.2.a.4 Footnote **	SR 3.8.1.7	----------NOTE---------- All DG starts may be preceded by an engine prelube period. ----------		
4.8.1.1.2.a.4 DOC L04		Verify each DG starts from standby condition and achieves: a. In ≤ ~~[~~10~~]~~ seconds, voltage ≥ ~~[3740]~~ V and frequency ≥ ~~[58.8]~~ Hz and (3950; 59.4) b. Steady state voltage ≥ ~~[3740]~~ V and ≤ ~~[4580]~~ V, and frequency ≥ ~~[58.8]~~ Hz and ≤ ~~[61.2]~~ Hz. (3950; 4370; 59.4; 60.6)	~~[184 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	3

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	~~NOTE~~ ~~[This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.]~~	
4.8.1.1.1.b	~~[~~ Verify ~~[~~automatic ~~[~~and~~]~~ manual~~]~~ transfer of AC power sources from the normal ~~offsite~~ circuit to ~~each alternate [required]~~ offsite circuit. (the)	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]]~~ (3) (1)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	----------NOTES---------- 1. [This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 2. If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ [0.9]. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable.] ----------	
	Verify each DG rejects a load greater than or equal to its associated single largest post-accident load and:	[[18] months OR
	a. Following load rejection, the frequency is ≤ [63] Hz, (66.7, 1.8) b. Within [3] seconds following load rejection, the voltage is ≥ [3740] V and ≤ [4580] V, and (1.8, 3950, 4370) c. Within [3] seconds following load rejection, the frequency is ≥ [58.8] Hz and ≤ [61.2] Hz. (59.4, 60.6)	In accordance with the Surveillance Frequency Control Program]

DOC M05
4.8.1.1.2.e.2 NOTE
DOC M05
4.8.1.1.2.e.2
DOC L08
3

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.1~~0~~ (9) ----------NOTES---------- 1. ~~[~~This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 2. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ ~~[~~0.9~~]~~. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable.~~]~~ ---------- Verify each DG does not trip, and voltage is maintained ≤ ~~[5000]~~ V during and following a load rejection of ≥ ~~[4500] kW and ≤ [5000]~~ kW. (4784) (3669)	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3) (6)

DOC M08

4.8.1.1.2.e.3 NOTE

DOC M08

4.8.1.1.2.e.3 DOC M10

Move SR to after ITS SR 3.8.1.18

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.1~~1~~ 0	----------NOTES----------	
4.8.1.1.2.e.4 Footnote ****	1. All DG starts may be preceded by an engine prelube period.	6
4.8.1.1.2.e.4 NOTE	2. This Surveillance shall not normally be performed in MODE 1, or 2~~, 3, or 4~~. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.	7

4.8.1.1.2.e.4	Verify on an actual or simulated loss of offsite power signal:	~~[[18] months~~
		~~OR~~
4.8.1.1.2.e.4.a	a. De-energization of emergency buses,	In accordance with the Surveillance Frequency Control Program ~~]~~
4.8.1.1.2.e.4.b	b. Load shedding from emergency buses,	
	c. DG auto-starts from standby condition and:	
	1. Energizes permanently connected loads in ≤ ~~[~~10~~]~~ seconds,	
	2. Energizes auto-connected ~~shutdown~~ emergency loads through ~~[automatic~~ load sequence~~r]~~ timers,	3 1
	3. Maintains steady state voltage ≥ ~~[3740]~~ 3950 V and ≤ ~~[4580]~~ 4370 V,	
	4. Maintains steady state frequency ≥ ~~[58.8]~~ 59.4 Hz and ≤ ~~[61.2]~~ 60.6 Hz, and	
	5. Supplies permanently connected ~~[~~and auto-connected~~]~~ ~~shutdown~~ emergency loads for ≥ 5 minutes.	1

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.1~~2~~ (1)	
----------NOTES----------	
1. ~~[~~ All DG starts may be preceded by an engine prelube period.	
2. This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.	

Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal each DG auto-starts from standby condition and:	~~[[18] months~~ ~~OR~~
a. In ≤ ~~[~~10~~]~~ seconds after auto-start and during tests, achieves voltage ≥ ~~[3740]~~ V and frequency ≥ ~~[58.8]~~ Hz, (3950, 59.4)	In accordance with the Surveillance Frequency Control Program~~]]~~
b. Achieves steady state voltage ≥ ~~[3740]~~ V and ≤ ~~[4580]~~ V and frequency ≥ ~~[58.8]~~ Hz and ≤ ~~[61.2]~~ Hz, (3950, 4370, 59.4, 60.6)	
c. Operates for ≥ 5 minutes,	
d. Permanently connected loads remain energized from the offsite power system, and	
e. Emergency loads are energized ~~[~~or auto-connected through the ~~automatic~~ load sequence~~r]~~ from the offsite power system. (timers)	

4.8.1.1.2.e.5 Footnote ****
4.8.1.1.2.e.5 NOTE
4.8.1.1.2.e.5 DOC A07
4.8.1.1.2.e.5.a DOC L04
4.8.1.1.2.e.5.c
4.8.1.1.2.e.5.b
DOC M06

(6) (3) (1)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.1~~8~~ (2) ----------NOTE---------- ~~[~~This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.~~]~~ ----------	
4.8.1.1.2.e.6 NOTE Verify each DG's noncritical automatic trips are bypassed on ~~[~~actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal~~]~~.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

(6)

4.8.1.1.2.e.6.c DOC L09

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.1~~4~~ (3) ----------NOTES---------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. ~~This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced.~~ Credit may be taken for unplanned events that satisfy this SR. 3. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ ~~[~~0.9~~]~~. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ----------	
Verify each DG operates for ≥ 24 hours: a. For ≥ ~~[~~2~~]~~ hours loaded ≥ ~~[5250]~~ 3800 kW and ≤ ~~[5500]~~ 3934 kW and b. For the remaining hours of the test loaded ≥ ~~[4500]~~ 3450 kW and ≤ ~~[5000]~~ 3669 kW.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

4.8.1.1.2.e.7 Footnote #

4.8.1.1.2.e.7 NOTE

DOC M05

4.8.1.1.2.e.7

DOC M09

DOC M10

(6) (R1) (1) (3) (3)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.~~15~~ (20) (Move SR to after ITS SR 3.8.1.19)	(8)
----------NOTES----------	
1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ~~≥ [2] hours~~ loaded ≥ ~~[4500]~~ (3450) kW and ≤ ~~[5000]~~ (3669) kW. Momentary transients outside of load range do not invalidate this test. (4.8.1.1.2.e.13 DOC A10; DOC M10; 4.8.1.1.2.e.13 Footnote #)	for ≥ 2 hours or until operating temperatures have stabilized. (1)
2. All DG starts may be preceded by an engine prelube period.	

Verify each DG starts and achieves: (4.8.1.1.2.e.13)	~~[[18] months~~
a. In ≤ ~~[~~10~~]~~ seconds, voltage ≥ ~~[3740]~~ (3950) V and frequency ≥ ~~[58.8]~~ (59.4) Hz and	~~OR~~
b. Steady state voltage ≥ ~~[3740]~~ (3950) V and ≤ ~~[4580]~~ (4370) V, and frequency ≥ ~~[58.8]~~ (59.4) Hz and ≤ ~~[61.2]~~ (60.6) Hz.	In accordance with the Surveillance Frequency Control Program~~]~~ (3)
SR 3.8.1.1~~6~~ (4)	(8)
----------NOTE---------- (4.8.1.1.2.e.9 NOTE)	
This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.	

Verify each DG: (4.8.1.1.2.e.9)	~~[[18] months~~
a. Synchronizes with offsite power source while loaded with emergency loads upon ~~a~~ (an actual or) simulated restoration of offsite power, (4.8.1.1.2.e.9.a)	~~OR~~
b. Transfers loads to offsite power source, and (4.8.1.1.2.e.9.b)	In accordance with the Surveillance Frequency Control Program~~]~~ (1) (3)
c. Returns to ready-to-load operation. (4.8.1.1.2.e.9.c)	

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.1~~7~~ (5) ----------NOTE---------- 4.8.1.1.2.e.10 NOTE ~~[~~This Surveillance shall not normally be performed in MODE 1, 2~~, 3, or 4~~. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- 4.8.1.1.2.e.10 Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to ready-to-load operation and ~~[~~b. Automatically energizing the emergency load from offsite power.~~]~~	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]]~~
SR 3.8.1.1~~8~~ (6) ----------NOTE---------- 4.8.1.1.2.e.12 NOTE ~~[~~This Surveillance shall not normally be performed in MODE 1, 2~~, 3, or 4~~. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.~~]~~ ---------- 4.8.1.1.2.e.12 Verify interval between each sequenced load block is within ± ~~[10%~~ of design interval~~]~~ for each ~~emergency [and shutdown]~~ load sequence~~r~~. 1 second timer	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.19	
----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, ~~3, or 4~~. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ----------	
Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal:	~~[~~ ~~[18] months~~ ~~OR~~
a. De-energization of emergency buses,	In accordance with the Surveillance Frequency Control Program ~~]~~
b. Load shedding from emergency buses,	
c. DG auto-starts from standby condition and:	
1. energizes permanently connected loads in ≤ ~~[~~10~~]~~ seconds,	
2. energizes auto-connected emergency loads through ~~[~~load sequence~~r]~~,	
3. achieves steady state voltage ≥ ~~[3740]~~ V and ≤ ~~[4580]~~ V,	
4. achieves steady state frequency ≥ ~~[58.8]~~ Hz and ≤ ~~[61.2]~~ Hz, and	
5. supplies permanently connected ~~[~~and auto-connected~~]~~ emergency loads for ≥ ~~[~~5~~]~~ minutes.	

7
8
4.8.1.1.2.e.6 Footnote ****
7
or
4.8.1.1.2.e.6 NOTE
4.8.1.1.2.e.6 DOC L09
4.8.1.1.2.e.6.a
4.8.1.1.2.e.6.a
4.8.1.1.2.e.6.b
3
2
timers
3950
4370
59.4
60.6

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.~~20~~ (18) ----------NOTE---------- All DG starts may be preceded by an engine prelube period. ----------	(8)
Verify, when started simultaneously from standby condition, each DG achieves: a. In ≤ ~~[~~10~~]~~ seconds, voltage ≥ ~~[3740]~~ (3950) V and frequency ≥ ~~[58.8]~~ (59.4) Hz and b. Steady state voltage ≥ ~~[3740]~~ (3950) V and ≤ ~~[4580]~~ (4370) V, and frequency ≥ ~~[58.8]~~ (59.4) Hz and ≤ ~~[61.2]~~ (60.6) Hz.	~~[10 years~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (3)

4.8.1.1.2.f Footnote ****

4.8.1.1.2.f

DOC M07
DOC L04
DOC L11
DOC L12

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.1, AC SOURCES - OPERATING

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. ISTS LCO 3.8.1.c and ISTS 3.8.1 ACTION F have been deleted since St. Lucie Plant (PSL) Units 1 and 2 do not use automatic load sequencers. Each load or load block is sequenced by the use of its associated time delay relay (i.e., load sequence timers). Each major ESF component has an individual time delay relay that operates individual components, instead of an electronic load sequencer that sequences the required ESF components on a bus. Thus, if a single time delay relay fails, only the affected component and the DG are impacted. ISTS SR 3.8.1.11 (ITS SR 3.8.1.10), ISTS SR 3.8.1.12 (ITS SR 3.8.1.11), ISTS SR 3.8.1.18 (ITS SR 3.8.1.16), and ISTS SR 3.8.1.19 (ITS SR 3.8.1.17) have been revised to reflect the use of load sequence timers.

3. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. Changes made to reflect the current licensing basis approved in PSL Unit 1 License Amendment 170 and PSL Unit 2 License Amendment 115.

5. ITS 3.8.1 ACTION F has been added to reflect CTS 3.8.1.1 Action d.2. ITS 3.8.1 Condition F applies when the Required Action and associated Completion Time of Condition C is not met with a Required Action to be in MODE 3 and a Completion Time of 6 hours. When two offsite circuits are inoperable, the original PSL Unit 1 and Unit 2 Technical Specifications only require the unit to be placed in Hot Standby (MODE 3) within 6 hours when one of two offsite circuits cannot be restored to OPERABLE status within 24 hours.

   When both offsite circuits are inoperable, the DGs are supplying power to the loads necessary to maintain the reactor shutdown. This change is acceptable because remaining in MODE 3 allows operators to focus on restoration of at least one offsite circuit to OPERABLE status instead of shifting their attention to a plant cooldown. In MODE 3, the steam generators continue to be available for steaming to maintain RCS parameters and steam generator makeup can be provided by the turbine driven auxiliary feedwater pump. This can minimize DG loading while attempts are made to restore offsite power. If one or more offsite circuits remains inoperable for > 72 hours while in MODE 3, the actions of Condition A (one offsite circuit inoperable) would not be met and Condition H would apply. Required Action H.2 will further require action to place the unit in MODE 4. Voluntary entry into MODE 4 or 5 is not restricted as these MODES are also acceptable low risk states. Concomitant changes are made to ISTS 3.8.1, Condition G.

6. ISTS SR 3.8.1.10 is not included in the Unit 1 PSL ITS. This SR is not included in the Unit 1 PSL current licensing basis and has not been retained in ITS. ISTS SR 3.8.1.10 requires verification that each DG will not trip and will maintain voltage within a maximum limit on a full load rejection test. A full load rejection test is not required by the CTS requirements. ITS retains the CTS requirement to

perform the largest post-accident load rejection test (SR 3.8.1.9). This change is acceptable because the DG output breaker current trip is greater than the current trip of the individual load breakers. Therefore, on a system fault, the breaker of the individual loads will trip before the DG output breaker trips. Tripping of the DG output breaker with all loads connected is the most likely way to cause a 100% load rejection. Therefore, testing of the largest post-accident load rejection is sufficient to ensure DG reliability. To maintain consistency between Unit 1 and Unit 2 SRs, ISTS SR 3.8.1.10 has been renumbered as SR 3.8.1.19 in the Unit 2 ITS. Subsequent ISTS Surveillances have been renumbered in the Unit 1 and Unit 2 ITS, as applicable.

7. ISTS SR 3.8.1.11, ISTS SR 3.8.1.17, ISTS SR 3.8.1.18 and ISTS SR 3.8.1.19 include a NOTE modifying the respective surveillance stating that the Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. These ISTS SRs are changed to state that the Surveillance shall not normally be performed in MODE 1 or 2, as approved in Unit 1 License Amendment 250 and Unit 2 License Amendment 202.

8. ISTS SR 3.8.1.15 is not included in the Unit 1 PSL ITS. ISTS SR 3.8.1.15 requires the hot fast re-start of each DG to rated voltage and frequency after operating for at least 2 hours. This test is normally associated with the requirement to perform a 24-hour run to establish the required “hot” conditions. ISTS SR 3.8.1.15 has not been included in the Unit 1 ITS since it is not consistent with current licensing basis and testing practices for the DGs. ISTS SR 3.8.1.15 tests a design feature (the hot restart capability of the DG) that is not credited in the PSL accident analysis. No postulated design basis event results in the need to restart the DG following a shutdown from hot conditions and the potential consequences of failure of DG to restart from hot condition are bounded by the potential consequences that would result from an immediate failure of the DG itself. Since the accident analysis assumes no operator actions associated with the Emergency Core Cooling System take place in the first 10 minutes following an accident, verification of this design feature does not materially contribute to the demonstration of DG OPERABILITY. To maintain consistency between Unit 1 and Unit 2 SRs, ISTS SR 3.8.1.15 has been renumbered as SR 3.8.1.20 in the Unit 2 ITS. Subsequent ISTS Surveillances have been renumbered in the Unit 1 and Unit 2 ITS, as applicable.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The unit Class 1E Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate(s)), and the onsite standby power sources (Train A and Train B diesel generators (DGs)). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single DG.

four

startup

Offsite power is supplied to the unit switchyard(s) from the transmission network by ~~[two]~~ transmission lines. From the switchyard(s), two electrically and physically separated circuits provide AC power, through ~~[~~step down ~~station auxiliary~~ transformers~~]~~, to the 4.16 kV ESF buses. A detailed description of the offsite power network and the circuits to the Class 1E ESF buses is found in the FSAR, Chapter ~~[~~8~~]~~ (Ref. 2).

2

Insert 1

1

U

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus or buses.

Insert 2

~~Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System. Within [1 minute] after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the load sequencer~~.

1

3

actuation signal

1B

1A3

1A

1B3

-low

4.16kV

The onsite standby power source for each 4.16 kV ESF bus is a dedicated DG. DGs ~~[1]~~ and ~~[2]~~ are dedicated to ESF buses ~~[1]~~ and ~~[2]~~, respectively. A DG starts automatically on a safety injection (SI) ~~signal~~ (i.e., low pressurizer pressure or high containment pressure signals) or on an ~~[~~ESF bus degraded voltage or undervoltage signal~~]~~. After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or

AS

2

1

St. Lucie – Unit 1

1

1

Insert 1

The circuit between each startup transformer and the associated 4.16 kV ESF bus includes two non-ESF buses: 2A4 and 1A2 buses for Train A; and 2B4 and 1B2 buses for Train B. Alternately, AC power can be supplied from a Unit 2 startup transformer; 2A or 2B via the non-ESF buses, by manually aligning the opposite unit startup transformer (i.e., removing and installing the startup transformer feeder breaker in a different breaker cubicle) to the unit 4.16 kV ESF bus.

1

Insert 2

Normal power for the onsite Class 1E AC Distribution System is supplied through the unit auxiliary transformers. If the normal onsite power source is lost to the onsite Class 1E AC Electrical Distribution System (i.e., main generator trip) an automatic fast transfer is initiated to transfer the 4.16 kV ESF buses from the unit auxiliary transformers to the startup transformers (offsite power). The transfer occurs in approximately three cycles.

BASES

BACKGROUND (continued)

degraded voltage, independent of or coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone. Following the trip of offsite power, [a sequencer/an undervoltage signal] strips nonpermanent loads from the ESF bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within [1] minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for Train A and Train B DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is [7000] kW with [10]% overload permissible for up to 2 hours in any 24 hour period. The ESF loads that are powered from the 4.16 kV ESF buses are listed in Reference 2.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the FSAR, Chapter [6] (Ref. 4) and Chapter [15] (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE during accident conditions in the event of:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. An assumed loss of all offsite power or all onsite AC power and

b. A worst case single failure.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System and separate and independent DGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

U

Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit. 1

~~[In addition, one required automatic load sequencer per train must be OPERABLE.]~~ 2

qualified physically independent, 1

Each offsite circuit must be capable of providing three phases of AC power, maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses.

Insert 3

~~[Offsite circuit #1 consists of Safeguards Transformer B, which is supplied from Switchyard Bus B, and is fed through breaker 52-3 powering the ESF transformer XNB01, which, in turn, powers the #1 ESF bus through its normal feeder breaker. Offsite circuit #2 consists of the Startup Transformer, which is normally fed from the Switchyard Bus A, and is fed through breaker PA 0201 powering the ESF transformer, which, in turn, powers the #2 ESF bus through its normal feeder breaker.]~~ 2

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This will be accomplished within ~~[~~10~~]~~ seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals~~,~~ and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode. 2 5

Insert 3 (1)

Each offsite circuit consists of a unit startup transformer (i.e., 1A, 1B) and the associated circuit path, including non-ESF buses (i.e., 2A4 and 1A2 buses; 2B4 and 1B2 buses), up to and including the feeder breaker to the respective unit 4.16 kV ESF bus (i.e., 1A3 bus, 1B3 bus). Alternately, when one unit startup transformer is unavailable, the associated offsite circuit may consist of an opposite unit startup transformer (i.e., 2A or 2B) and the associated circuit path, including non-ESF buses, up to and including the feeder breaker to the respective unit 4.16 kV ESF bus.

BASES

LCO (continued)

Proper sequencing of loads, ~~[~~including tripping of nonessential loads,~~]~~ is a required function for DG OPERABILITY. 2

The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the DGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical. ~~A circuit may be connected to more than one ESF bus, with fast transfer capability to the other circuit OPERABLE, and not violate separation criteria.~~ A circuit that is not connected to an ESF bus is required to have OPERABLE fast transfer interlock mechanisms to at least ~~two~~ ESF bus~~es~~ to support OPERABILITY of that circuit. 1

one

Insert 4

APPLICABILITY

The AC sources ~~[and sequencers]~~ are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that: 3 2

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable DG. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable DG and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

To ensure a highly reliable power source remains with the one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1

1

Insert 4

Each startup transformer set (1A-2A, 1B-2B) is provided with a manual switching arrangement that permits powering the associated Class 1E electrical distribution subsystem from either unit's startup transformer. A single startup transformer is adequate to accommodate the emergency and auxiliary loads of a unit during a postulated DBA. Therefore, when utilized as a qualified offsite AC source to Unit 1, the Unit 2 startup transformer may be aligned to ESF buses of both units and not violate the criterion for shared systems between units provided the Unit 2 startup transformer is not supplying power to a Unit 2 bus.

BASES

ACTIONS (continued)

acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

~~REVIEWER'S NOTE~~

~~The turbine driven auxiliary feedwater pump is only required to be considered a redundant required feature, and, therefore, required to be determined OPERABLE by this Required Action, if the design is such that the remaining OPERABLE motor or turbine driven auxiliary feedwater pump(s) is not by itself capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.~~

4

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features. These features are powered from the redundant AC electrical power train. This includes motor and steam driven auxiliary feedwater pumps~~. Single train systems, such as turbine driven auxiliary feedwater pumps, may not be included~~.

The steam driven auxiliary feedwater pump is only considered a redundant required feature in MODES 1, 2, and 3 because it is not required to be OPERABLE in MODE 4.

1

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train has no offsite power supplying its loads and

b. A required feature on the other train is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

BASES

ACTIONS (continued)

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System. 2

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

BASES

ACTIONS (continued)

~~REVIEWER'S NOTE~~

~~The turbine driven auxiliary feedwater pump is only required to be considered a redundant required feature, and, therefore, required to be determined OPERABLE by this Required Action, if the design is such that the remaining OPERABLE motor or turbine driven auxiliary feedwater pump(s) is not by itself capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.~~

4

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. This includes motor driven auxiliary feedwater pumps~~. Single train systems, such as turbine driven auxiliary feedwater pumps, are not included~~. Redundant required feature failures consist of inoperable features with a train, redundant to the train that has an inoperable DG.

and steam

1

The steam driven auxiliary feedwater pump is only considered a redundant required feature in MODES 1, 2, and 3 because it is not required to be OPERABLE in MODE 4.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists and

b. A required feature on the other train is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

St. Lucie – Unit 1

1

BASES

ACTIONS (continued)

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently, is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DG~~(s)~~, the other DG~~(s)~~ would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG~~(s)~~, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

For example: if the DG inoperability is due to an inoperable support system, an independently testable component, preplanned preventative maintenance or testing.

the

1

1

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the ~~[~~plant corrective action program~~]~~ will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

2

9

According to Generic Letter 84-15 (Ref. ~~7~~), ~~[~~24~~]~~ hours is reasonable to confirm that the OPERABLE DG~~(s)~~ is not affected by the same problem as the inoperable DG.

1 2

1

BASES

ACTIONS (continued)

B.4

Insert 5

~~According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition B for a period that should not exceed 72 hours.~~ 1

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The ~~72-hour~~ Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~

14 day 1

2

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. This includes motor driven auxiliary feedwater pumps. ~~Single train features, such as turbine driven auxiliary pumps, are not included in the list.~~

and steam 1

The steam driven auxiliary feedwater pump is only considered a redundant required feature in MODES 1, 2, and 3 because it is not required to be OPERABLE in MODE 4.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable and

b. A required feature is inoperable.

Insert 5 (1)

The 14-day Completion Time is based on the findings of a deterministic and probabilistic safety analysis. The risk impact of the Completion Time has been evaluated pursuant to the risk assessment and management provisions of the Maintenance Rule, 10 CFR 50.65(a)(4), and the associated implementation guidance, Regulatory Guide 1.160. Per NRC Letter, "St. Lucie Unit 1 – Issuance of Amendment Regarding Diesel Generator Allowed Outage Time Extension (TAC No. MA7205)," dated January 19, 2001, (Ref. 7) operation may continue in Condition B for a period of 14 days provided compensatory measures are taken.

BASES

ACTIONS (continued)

If at any time during the existence of Condition C (two offsite circuits inoperable) and a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ 2

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

BASES

ACTIONS (continued)

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems - Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 12 hours. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ 2

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ 2

E.1

With Train A and Train B DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued

BASES

ACTIONS (continued)

operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Regulatory Guide 1.93 (Ref. 6), with both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

Insert 6

~~[F.1~~

~~The sequencer(s) is an essential support system to [both the offsite circuit and the DG associated with a given ESF bus]. [Furthermore, the sequencer is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus.] Therefore, loss of an [ESF bus sequencer] affects every major ESF system in the [division]. The [12] hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer OPERABILITY. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.] This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods when the sequencer is inoperable is minimal.~~

~~This Condition is preceded by a Note that allows the Condition to be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads under any conditions. Implicit in this Note is the concept that the Condition must be retained if any sequencer failure mode results in the inability to start all or part of the safety loads when required, regardless of power availability, or results in overloading the offsite power circuit to a safety bus during an event, thereby causing its failure. Also implicit in the Note, is that the Condition is not applicable to any train that does not have a sequencer.]~~

1

2

G.1 and G.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

4

St. Lucie – Unit 1

1

1

Insert 6

If a Required Action of Condition C cannot be performed within its associated Completion Time, the unit must be brought to a MODE in which plant risk is minimized. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

When both offsite circuits are inoperable, the DGs are supplying power to the loads necessary to maintain the reactor shutdown. Remaining in MODE 3 allows operators to focus on restoration of at least one offsite circuit to OPERABLE status instead of shifting their attention to a plant cooldown. In MODE 3, the steam generators continue to be available for steaming to maintain RCS parameters and steam generator makeup can be provided by the turbine driven auxiliary feedwater pump. This can minimize DG loading while attempts are made to restore offsite power. If one or more offsite circuits remains inoperable for > 72 hours while in MODE 3, the actions of Condition A would not be met and Condition H would apply. Required Action H.2 will further require action to place the unit in MODE 4. However, voluntary entry into MODE 4 or 5 may be made as these MODES are also acceptable low risk states.

The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

4

~~If the inoperable AC electrical power sources cannot be restored to OPERABLE status within the required Completion Time,~~ the unit must be brought to a MODE in which plant risk is minimized. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

1

If a Required Action of Conditions A, B, D, or E cannot be performed within its associated Completion Time,

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~7~~). In MODE 5, it is likely that increased plant maintenance activities (particularly those involving the switchyard) will make the plant more susceptible to loss of offsite power events. In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

8

1

Required Action G.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

St. Lucie – Unit 1

1

BASES

ACTIONS (continued)

H.1

Condition H corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. ~~9~~ 10).

Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 1~~0~~), and Regulatory Guide 1.137 (Ref. 1~~1~~), as addressed in the FSAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of ~~[3740] V is 90% of the nominal 4160 V output voltage. This value, which is specified in ANSI C84.1-1982 (Ref. 12), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 80% of name plate rating. The specified maximum steady state output voltage of [4756] V is equal to the maximum operating voltage specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).~~

SR 3.8.1.1

This SR ensures correct breaker alignment for each ~~[required]~~ offsite circuit to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The SR also verifies the indicated availability of three-phase AC electrical power from each ~~[required]~~ offsite circuit to the

Insert 7 (1)

3950 V is 95% of the nominal 4160 V output voltage. This value, which is conservative with respect to the value specified in ANSI C84.1-1989 (Ref. 13), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level. The minimum steady state output voltage ensures the operating system voltage for the 480 V switchgear is ≥ 90% of nominal bus voltage, which bounds the 90% criterion of the safety related motor voltage ratings. The specified maximum steady state output voltage of 4370 V bounds the maximum operating bus voltage. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is such that continuous operation of motor loads does not result in a failure of the load due to overheating. The specified minimum and maximum frequencies of the DG are 59.4 Hz and 60.6 Hz, respectively. These values are equal to ± 1% of the 60 Hz nominal frequency and are conservative with respect to the recommendations given in Regulatory Guide 1.9 (Ref. 3). The under frequency limit considers reduced pump flow and discharge pressure and reduced motor operated valve stroke times during accident sequences. The over frequency limit considers net positive suction head of safety related pumps, increased differential pressure across closed valves, and increased DG fuel oil and lube oil consumption.

BASES

SURVEILLANCE REQUIREMENTS (continued)

onsite distribution network. [The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 1 for SR 3.8.1.2 and Note for SR 3.8.1.7) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading by an engine prelube period.

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the DGs are started from standby conditions. Standby conditions for a DG mean the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

[In order to reduce stress and wear on diesel engines, the DG manufacturers recommend a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. This is the intent of Note 2, which is only applicable when such modified start procedures are recommended by the manufacturer.

SR 3.8.1.7 requires that the DG starts from standby conditions and achieves required voltage and frequency within 10 seconds. The 10 second start requirement supports the assumptions of the design basis LOCA analysis in the FSAR, Chapter [15] (Ref. 5).]

BASES

SURVEILLANCE REQUIREMENTS (continued)

The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 2) when a modified start procedure as described above is used. If a modified start is not used, 10 second start requirement of SR 3.8.1.7 applies.

Since SR 3.8.1.7 requires a 10 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2.

In addition to the SR requirements, the time for the DG to reach steady state operation, unless the modified DG start method is employed, is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

~~[The 31 day Frequency for SR 3.8.1.2 is consistent with Regulatory Guide 1.9 (Ref. 3). The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 8). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between ~~[~~0.8 lagging~~]~~ and ~~[~~1.0~~]~~. The 0.8 value is the design rating of the machine, while ~~[~~1.0~~]~~ is an operational limitation ~~[~~to ensure circulating currents are minimized~~]~~. ~~[The 31 day Frequency for this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 3).~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit will not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

engine mounted day tanks (one tank per engine)

This SR provides verification that the level of fuel oil in the ~~day tank [and engine mounted tank]~~ is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil ~~for a minimum of 1 hour of DG operation at full load plus 10%~~.

to operate the engine at rated capacity for a sufficient duration to allow maintenance or valve realignment

~~[The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day ~~[and engine mounted]~~ tanks eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. ~~[The Surveillance Frequency of 31 days is established by Regulatory Guide 1.137 (Ref. 11). This SR is for preventive maintenance.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed during the performance of this Surveillance.

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from ~~its associated storage tank~~ the fuel oil storage system to its associated day tank~~s~~. This is required to support continuous operation of

BASES

SURVEILLANCE REQUIREMENTS (continued)

transfer system

standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping ~~system~~ is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE. 1

~~[[The Frequency for this SR is variable, depending on individual system design, with up to a [92] day interval. The [92] day Frequency corresponds to the testing requirements for pumps as contained in the ASME Code (Ref. 13); however, the design of fuel transfer systems is such that pumps will operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day [and engine mounted] tanks during or following DG testing. In such a case, a 31 day Frequency is appropriate. Since proper operation of fuel transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]]~~ 4 2

SR 3.8.1.7

See SR 3.8.1.2.

~~[~~ SR 3.8.1.8

, (i.e., unit auxiliary transformer)

offsite

Transfer of each ~~[~~4.16 kV ESF bus~~]~~ power supply from the normal ~~offsite~~ circuit to the ~~alternate~~ offsite circuit demonstrates the OPERABILITY of the ~~alternate~~ circuit distribution network to power the shutdown loads. 1

~~[The [18 month] Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR .]~~

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. ~~[~~For this unit, the single load for each DG and its horsepower rating is ~~as follows:]~~. This Surveillance may be accomplished by:

an intake cooling water pump with a nameplate rating of 600 hp

BASES

SURVEILLANCE REQUIREMENTS (continued)

a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power or while solely supplying the bus, or

b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

As required by IEEE-308 (Ref. 14), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The ~~[3]~~ seconds specified is equal to 60% of a ~~typical 5~~ second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the ~~system~~ must recover following load rejection~~. [The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 10).~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

This SR is modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns)

BASES

SURVEILLANCE REQUIREMENTS (continued)

provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR. Note 2 ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ ~~[~~0.9~~]~~. This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, Note 2 allows the Surveillance to be conducted at a power factor other than ≤ ~~[~~0.9~~]~~. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ ~~[~~0.9~~]~~ results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to ~~[~~0.9~~]~~ while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of ~~[~~0.9~~]~~ may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close as practicable to ~~[~~0.9~~]~~ without exceeding the DG excitation limits. 2

~~REVIEWER'S NOTE~~

~~The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable~~:

~~a. Performance of the SR will not render any safety system or component inoperable,~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and~~

~~c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.10

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG will not trip upon loss of the load. These acceptance criteria provide DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

[The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 10) and is intended to be consistent with expected fuel cycle lengths.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

This SR is modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbation to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system

BASES

SURVEILLANCE REQUIREMENTS (continued)

when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR. Note 2 ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ [0.9]. This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, Note 2 allows the Surveillance to be conducted at a power factor other than ≤ [0.9]. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ [0.9] results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to [0.9] while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of [0.9] may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close as practicable to [0.9] without exceeding the DG excitation limits.

1

---------- REVIEWER'S NOTE ----------

The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

a. Performance of the SR will not render any safety system or component inoperable,

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and

c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.

4

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.1~~1~~

As required by Regulatory Guide 1.108 (Ref. 1~~0~~), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG auto-start time of ~~[~~10~~]~~ seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, high pressure injection systems are not capable of being operated at full flow, or shutdown cooling (SDC) systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

~~[The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

~~[~~SR 3.8.1.1~~2~~

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (~~[~~10~~]~~ seconds) from the ~~design basis~~ actuation signal ~~(LOCA signal)~~ and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.1~~2~~.d and SR 3.8.1.1~~2~~.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The requirement to verify the connection of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, high pressure injection systems are not capable of being operated at full flow, or SDC systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

~~[The Frequency of [18 months] takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~]~~ 2

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing

BASES

SURVEILLANCE REQUIREMENTS (continued)

OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.~~]~~ 2

SR 3.8.1.1~~3~~ 2 6

This Surveillance demonstrates that DG noncritical protective functions (e.g., high jacket water temperature) are bypassed on a loss of voltage signal concurrent with an ESF actuation ~~test~~ signal. Noncritical automatic trips are all automatic trips except: 1

a. Engine overspeed; and 1

b. Generator differential current~~;~~

~~[c. Low lube oil pressure;~~

~~d. High crankcase pressure; and~~ 2

~~e. Start failure relay.]~~

The noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

~~[The [18 month] Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 2

The SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

~~---------- REVIEWER'S NOTE ----------~~
~~The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

~~a. Performance of the SR will not render any safety system or component inoperable,~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.~~

SR 3.8.1.1~~4~~

Regulatory Guide 1.108 (Ref. 1~~0~~), paragraph 2.a.(3), requires demonstration that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ~~≥ [~~2~~]~~ hours of which is at a load equivalent to 110% of the continuous duty rating) and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

the 2 hour rating (i.e.,

This SR meets the intent of this requirement to the extent practical by loading the DG to approximately 109% of the generator continuous duty rating not to exceed the 4 hour rating. This loading requirement challenges the full loading capability of the DG without unnecessarily overloading the DG.

The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

~~[The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance is modified by three Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. ~~The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.~~ 2

Note 2 provides an allowance that c

~~C~~redit may be taken for unplanned events that satisfy this SR. Note 3 ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ ~~[~~0.9~~]~~. This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, Note 3 allows the Surveillance to be conducted at a power factor other than ≤ ~~[~~0.9~~]~~. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ ~~[~~0.9~~]~~ results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to ~~[~~0.9~~]~~ while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of ~~[~~0.9~~]~~ may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close practicable to ~~[~~0.9~~]~~ without exceeding the DG excitation limits. 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.8.1.15~~

~~This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within [10] seconds. The [10] second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. [The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(5).~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least [2] hours at full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.~~

SR 3.8.1.1~~6~~

As required by Regulatory Guide 1.108 (Ref. 1~~0~~), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and that the DG can be returned to ready to load status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the DG to reload if a

BASES

SURVEILLANCE REQUIREMENTS (continued)

subsequent loss of offsite power occurs. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open and can receive and autoclose signal on bus undervoltage, and the load sequence timers are reset. 5

~~[The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~]~~ 2

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[~~SR 3.8.1.1~~7~~

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to ready to load operation if a ~~LOCA~~ actuation signal is received during operation in the test mode. Ready to load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 14), paragraph 6.2.6(2).

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.1~~2~~. The intent in the requirement associated with SR 3.8.1.1~~7~~.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

~~[The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(8); takes into consideration unit conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing

BASES

SURVEILLANCE REQUIREMENTS (continued)

following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR. ~~]~~ 2

SR 3.8.1.1~~8~~ 6 6

Under accident ~~[~~and loss of offsite power~~]~~ conditions loads are sequentially connected to the bus by the ~~[~~automatic load sequence~~r]~~. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The ~~[10]%~~ load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. ~~Reference 1 provides a summary of the automatic loading of ESF buses.~~ timers ±1 second 3 1

~~[The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(2); takes into consideration unit conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

~~REVIEWER'S NOTE~~

~~The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

~~a. Performance of the SR will not render any safety system or component inoperable,~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and~~

~~c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.~~

4

7

SR 3.8.1.1~~9~~

6

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.1~~1~~, during a loss of offsite power actuation ~~test~~ signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

~~[The Frequency of [18 months] takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of [18 months].~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they

BASES

SURVEILLANCE REQUIREMENTS (continued)

are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.~~20~~

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.

~~[The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10).~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated, and temperature maintained consistent with manufacturer recommendations.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter ~~[~~8~~]~~.
3. Regulatory Guide 1.9, Rev. ~~[~~3~~]~~.
4. FSAR, Chapter ~~[~~6~~]~~.

BASES

REFERENCES (continued)

5. FSAR, Chapter ~~[~~15~~]~~.

6. Regulatory Guide 1.93, Rev. ~~[ ], [date]~~.

~~7~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

~~8~~. Generic Letter 84-15.

~~9~~. 10 CFR 50, Appendix A, GDC 18.

~~10~~. Regulatory Guide 1.108, Rev. ~~[~~1~~], [August 1977]~~.

~~11~~. Regulatory Guide 1.137, Rev. ~~[ ], [date]~~.

~~12~~. ANSI C84.1-198~~2~~.

~~13~~. ~~ASME Code for Operation and Maintenance of Nuclear Power Plants.~~

14. IEEE Standard 308-~~[~~1978~~]~~.

7. NRC Letter, "St. Lucie Unit 1 – Issuance of Amendment Regarding Diesel Generator Allowed Outage Time Extension (TAC No. MA7205)," dated January 19, 2001 (ML010250042).

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The unit Class 1E Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate(s)), and the onsite standby power sources (Train A and Train B diesel generators (DGs)). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single DG.

Offsite power is supplied to the unit switchyard(s) from the transmission network by ~~[two]~~ four transmission lines. From the switchyard(s), two electrically and physically separated circuits provide AC power, through ~~[~~step down ~~station auxiliary~~ startup transformers~~]~~, to the 4.16 kV ESF buses. A detailed description of the offsite power network and the circuits to the Class 1E ESF buses is found in the FSAR, Chapter ~~[~~8~~]~~ U (Ref. 2). Insert 1 2 1

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus or buses.

Insert 2

~~Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System. Within [1 minute] after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the load sequencer.~~ 1

The onsite standby power source for each 4.16 kV ESF bus is a dedicated DG. DGs ~~[1]~~ 2A and ~~[2]~~ 2B are dedicated to ESF buses ~~[1]~~ 2A3 and ~~[2]~~ 2B3, respectively. A DG starts automatically on a safety injection (SI) ~~signal~~ -low (i.e., low pressurizer pressure or high containment pressure signals) or on an ~~[~~ESF~~]~~ 4.16kV bus degraded voltage or undervoltage ~~signal~~ actuation signal AS. After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or 3 1 2 1

1

Insert 1

The circuit between each startup transformer and the associated 4.16 kV ESF bus includes two non-ESF buses: 2A4 and 2A2 buses for Train A; and 2B4 and 2B2 buses for Train B. Alternately, AC power can be supplied from a Unit 1 startup transformer, 1A or 1B via the non-ESF buses, by manually aligning the opposite unit startup transformer (i.e., removing and installing the startup transformer feeder breaker in a different breaker cubicle) to the unit 4.16 kV ESF bus.

1

Insert 2

Normal power for the onsite Class 1E AC Distribution System is supplied through the unit auxiliary transformers. If the normal onsite power source is lost to the onsite Class 1E AC Electrical Distribution System (i.e., main generator trip) an automatic fast transfer is initiated to transfer the 4.16 kV ESF buses from the unit auxiliary transformers to the startup transformers (offsite power). The transfer occurs in approximately three cycles.

BASES

BACKGROUND (continued)

degraded voltage, independent of or coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone. Following the trip of offsite power, ~~[a sequencer/an undervoltage signal] strips~~ nonpermanent loads from the ESF bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus ~~by the automatic load sequencer~~. ~~The sequencing logic controls the permissive and starting signals to motor breakers~~to prevent overloading the DG by automatic load application.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within ~~[~~1~~]~~ minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for Train A and Train B DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is ~~[7000]~~ kW with ~~[10]% overload permissible for up to 2 hours in any 24 hour period~~. The ESF loads that are powered from the 4.16 kV ESF buses are listed in Reference 2.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 4) and Chapter ~~[~~15~~]~~ (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE during accident conditions in the event of:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. An assumed loss of all offsite power or all onsite AC power and

b. A worst case single failure.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System and separate and independent DGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Qualified offsite circuits are those that are described in the U FSAR and are part of the licensing basis for the unit. (1)

~~[In addition, one required automatic load sequencer per train must be OPERABLE.]~~ (2)

Each qualified offsite circuit must be physically independent, capable of providing three phases of AC power, maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses. (1)

Insert 3

~~[Offsite circuit #1 consists of Safeguards Transformer B, which is supplied from Switchyard Bus B, and is fed through breaker 52-3 powering the ESF transformer XNB01, which, in turn, powers the #1 ESF bus through its normal feeder breaker. Offsite circuit #2 consists of the Startup Transformer, which is normally fed from the Switchyard Bus A, and is fed through breaker PA 0201 powering the ESF transformer, which, in turn, powers the #2 ESF bus through its normal feeder breaker.]~~ (2)

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This will be accomplished within ~~[~~10~~]~~ seconds. (2) Each DG must also be capable of accepting required loads within the assumed loading sequence intervals~~,~~ and continue to operate until offsite power can be restored to the ESF buses. (5) These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode.

Insert 3 (1)

Each offsite circuit consists of a unit startup transformer (i.e., 2A, 2B) and the associated circuit path, including non-ESF buses (i.e., 2A4 and 2A2 buses; 2B4 and 2B2 buses), up to and including the feeder breaker to the respective unit 4.16 kV ESF bus (i.e., 2A3 bus, 2B3 bus). Alternately, when one unit startup transformer is unavailable, the associated offsite circuit may consist of an opposite unit startup transformer (i.e., 1A or 1B) and the associated circuit path, including non-ESF buses, up to and including the feeder breaker to the respective unit 4.16 kV ESF bus.

BASES

LCO (continued)

Proper sequencing of loads, ~~[~~including tripping of nonessential loads,~~]~~ is a required function for DG OPERABILITY. 2

The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the DGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical. ~~A circuit may be connected to more than one ESF bus, with fast transfer capability to the other circuit OPERABLE, and not violate separation criteria.~~ A circuit that is not connected to an ESF bus is required to have OPERABLE fast transfer interlock mechanisms to at least ~~two~~ ESF bus~~es~~ to support OPERABILITY of that circuit. 1

one

Insert 4

APPLICABILITY

The AC sources ~~[and sequencers]~~ are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that: 3 2

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable DG. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable DG and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

To ensure a highly reliable power source remains with the one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1

1

Insert 4

Each startup transformer set (1A-2A, 1B-2B) is provided with a manual switching arrangement that permits powering the associated Class 1E electrical distribution subsystem from either unit's startup transformer. A single startup transformer is adequate to accommodate the emergency and auxiliary loads of a unit during a postulated DBA. Therefore, when utilized as a qualified offsite AC source to Unit 2, the Unit 1 startup transformer may be aligned to ESF buses of both units and not violate the shared system criterion between units provided the Unit 1 startup transformer is not supplying power to a Unit 1 bus.

BASES

ACTIONS (continued)

acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

~~REVIEWER'S NOTE~~

~~The turbine driven auxiliary feedwater pump is only required to be considered a redundant required feature, and, therefore, required to be determined OPERABLE by this Required Action, if the design is such that the remaining OPERABLE motor or turbine driven auxiliary feedwater pump(s) is not by itself capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.~~

4

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features. These features are powered from the redundant AC electrical power train. This includes motor and steam driven auxiliary feedwater pumps~~. Single train systems, such as turbine driven auxiliary feedwater pumps, may not be included~~.

The steam driven auxiliary feedwater pump is only considered a redundant required feature in MODES 1, 2, and 3 because it is not required to be OPERABLE in MODE 4.

1

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train has no offsite power supplying its loads and

b. A required feature on the other train is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

BASES

ACTIONS (continued)

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System. 2

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

BASES

ACTIONS (continued)

~~REVIEWER'S NOTE~~

~~The turbine driven auxiliary feedwater pump is only required to be considered a redundant required feature, and, therefore, required to be determined OPERABLE by this Required Action, if the design is such that the remaining OPERABLE motor or turbine driven auxiliary feedwater pump(s) is not by itself capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.~~

4

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. This includes motor driven auxiliary feedwater pumps~~. Single train systems, such as turbine driven auxiliary feedwater pumps, are not included~~. Redundant required feature failures consist of inoperable features with a train, redundant to the train that has an inoperable DG.

and steam

1

The steam driven auxiliary feedwater pump is only considered a redundant required feature in MODES 1, 2, and 3 because it is not required to be OPERABLE in MODE 4.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists and

b. A required feature on the other train is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

BASES

ACTIONS (continued)

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently, is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DG~~(s)~~, the other DG~~(s)~~ would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG~~(s)~~, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

For example: if the DG inoperability is due to an inoperable support system, an independently testable component, preplanned preventative maintenance or testing.

the

1

1

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the ~~[~~plant corrective action program~~]~~ will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

2

9

According to Generic Letter 84-15 (Ref. ~~7~~), ~~[~~24~~]~~ hours is reasonable to confirm that the OPERABLE DG~~(s)~~ is not affected by the same problem as the inoperable DG.

2

1

St. Lucie – Unit 2

1

BASES

ACTIONS (continued)

B.4

Insert 5

~~According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition B for a period that should not exceed 72 hours.~~ 1

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The ~~72-hour~~ Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~

14 day 1

2

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. This includes motor driven auxiliary feedwater pumps. ~~Single train features, such as turbine driven auxiliary pumps, are not included in the list.~~

and steam 1

The steam driven auxiliary feedwater pump is only considered a redundant required feature in MODES 1, 2, and 3 because it is not required to be OPERABLE in MODE 4.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable and

b. A required feature is inoperable.

Insert 5 (1)

The 14-day Completion Time is based on the findings of a deterministic and probabilistic safety analysis. The risk impact of the Completion Time has been evaluated pursuant to the risk assessment and management provisions of the Maintenance Rule, 10 CFR 50.65(a)(4), and the associated implementation guidance, Regulatory Guide 1.160. Per NRC Letter, "St. Lucie Unit No. 2 – Issuance of Amendment Regarding Diesel Generator Allowed Outage Time Extension (TAC No. MA7206)," dated April 26, 2001, (Ref. 7) operation may continue in Condition B for a period of 14 days provided compensatory measures are taken.

BASES

ACTIONS (continued)

If at any time during the existence of Condition C (two offsite circuits inoperable) and a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ (2)

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

BASES

ACTIONS (continued)

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems - Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 12 hours. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ 2

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~ 2

E.1

With Train A and Train B DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued

BASES

ACTIONS (continued)

operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Regulatory Guide 1.93 (Ref. 6), with both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

Insert 6

~~[F.1~~

~~The sequencer(s) is an essential support system to [both the offsite circuit and the DG associated with a given ESF bus]. [Furthermore, the sequencer is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus.] Therefore, loss of an [ESF bus sequencer] affects every major ESF system in the [division]. The [12] hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer OPERABILITY. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.] This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods when the sequencer is inoperable is minimal.~~

~~This Condition is preceded by a Note that allows the Condition to be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads under any conditions. Implicit in this Note is the concept that the Condition must be retained if any sequencer failure mode results in the inability to start all or part of the safety loads when required, regardless of power availability, or results in overloading the offsite power circuit to a safety bus during an event, thereby causing its failure. Also implicit in the Note, is that the Condition is not applicable to any train that does not have a sequencer.]~~

1
2

G.1 and G.2

~~---------- REVIEWER'S NOTE ----------~~
~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

4

Insert 6 (1)

If a Required Action of Condition C cannot be performed within its associated Completion Time, the unit must be brought to a MODE in which plant risk is minimized. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

When both offsite circuits are inoperable, the DGs are supplying power to the loads necessary to maintain the reactor shutdown. Remaining in MODE 3 allows operators to focus on restoration of at least one offsite circuit to OPERABLE status instead of shifting their attention to a plant cooldown. In MODE 3, the steam generators continue to be available for steaming to maintain RCS parameters and steam generator makeup can be provided by the turbine driven auxiliary feedwater pump. This can minimize DG loading while attempts are made to restore offsite power. If one or more offsite circuits remains inoperable for > 72 hours while in MODE 3, the actions of Condition A would not be met and Condition H would apply. Required Action H.2 will further require action to place the unit in MODE 4. However, voluntary entry into MODE 4 or 5 may be made as these MODES are also acceptable low risk states.

The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~ 4

~~If the inoperable AC electrical power sources cannot be restored to OPERABLE status within the required Completion Time,~~ the unit must be brought to a MODE in which plant risk is minimized. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. 1

If a Required Action of Conditions A, B, D, or E cannot be performed within its associated Completion Time,

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. ~~7~~ 8). In MODE 5, it is likely that increased plant maintenance activities (particularly those involving the switchyard) will make the plant more susceptible to loss of offsite power events. In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. 1

Required Action G.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

H.1

Condition H corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. ~~9~~).

Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 1~~0~~), and Regulatory Guide 1.137 (Ref. 1~~1~~), as addressed in the FSAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of ~~[3740] V is 90% of the nominal 4160 V output voltage. This value, which is specified in ANSI C84.1-1982 (Ref. 12), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 80% of name plate rating. The specified maximum steady state output voltage of [4756] V is equal to the maximum operating voltage specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3)~~.

SR 3.8.1.1

This SR ensures correct breaker alignment for each ~~[required]~~ offsite circuit to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The SR also verifies the indicated availability of three-phase AC electrical power from each ~~[required]~~ offsite circuit to the

Insert 7 (1)

3950 V is 95% of the nominal 4160 V output voltage. This value, which is conservative with respect to the value specified in ANSI C84.1-1989 (Ref. 13), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level. The minimum steady state output voltage ensures the operating system voltage for the 480 V switchgear is ≥ 90% of nominal bus voltage, which bounds the 90% criterion of the safety related motor voltage ratings. The specified maximum steady state output voltage of 4370 V bounds the maximum operating bus voltage. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is such that continuous operation of motor loads does not result in a failure of the load due to overheating. The specified minimum and maximum frequencies of the DG are 59.4 Hz and 60.6 Hz, respectively. These values are equal to ± 1% of the 60 Hz nominal frequency and are conservative with respect to the recommendations given in Regulatory Guide 1.9 (Ref. 3). The under frequency limit considers reduced pump flow and discharge pressure and reduced motor operated valve stroke times during accident sequences. The over frequency limit considers net positive suction head of safety related pumps, increased differential pressure across closed valves, and increased DG fuel oil and lube oil consumption.

BASES

SURVEILLANCE REQUIREMENTS (continued)

onsite distribution network. [The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room. 2

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement. 4
----------] 2

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 1 for SR 3.8.1.2 and Note for SR 3.8.1.7) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading by an engine prelube period. 1

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the DGs are started from standby conditions. Standby conditions for a DG mean the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

[In order to reduce stress and wear on diesel engines, the DG manufacturers recommend a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. This is the intent of Note 2, which is only applicable when such modified start procedures are recommended by the manufacturer. 2

s

SR 3.8.1.2

SR 3.8.1.7 requires that the DG starts from standby conditions and achieves required voltage and frequency within 10 seconds. The 10 second start requirement supports the assumptions of the design basis LOCA analysis in the FSAR, Chapter [15] (Ref. 5).]

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Chapter 6 (Ref. 4) and

BASES

SURVEILLANCE REQUIREMENTS (continued)

The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 2) when a modified start procedure as described above is used. If a modified start is not used, 10 second start requirement of SR 3.8.1.7 applies.

Since SR 3.8.1.7 requires a 10 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2.

In addition to the SR requirements, the time for the DG to reach steady state operation, unless the modified DG start method is employed, is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

~~[The 31 day Frequency for SR 3.8.1.2 is consistent with Regulatory Guide 1.9 (Ref. 3). The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 8). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between ~~[~~0.8 lagging~~]~~ and ~~[~~1.0~~]~~. The 0.8 value is the design rating of the machine, while ~~[~~1.0~~]~~ is an operational limitation ~~[~~to ensure circulating currents are minimized~~]~~. ~~[The 31 day Frequency for this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 3).~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 2

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit will not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

to operate the engine at rated capacity for a sufficient duration to prevent cycling of the fuel oil transfer pump and maintain 60 minutes of operation at rated capacity.

SR 3.8.1.4

engine mounted day tanks (one tank per engine)

This SR provides verification that the level of fuel oil in the day tank ~~[and engine mounted tank]~~ is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil ~~for a minimum of 1 hour of DG operation at full load plus 10%~~. 2 1

~~[The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day ~~[and engine mounted]~~ tanks eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. ~~[The Surveillance Frequency of 31 days is established by Regulatory Guide 1.137 (Ref. 11). This SR is for preventive maintenance.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed during the performance of this Surveillance.

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of

s

BASES

SURVEILLANCE REQUIREMENTS (continued)

standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

[[The Frequency for this SR is variable, depending on individual system design, with up to a [92] day interval. The [92] day Frequency corresponds to the testing requirements for pumps as contained in the ASME Code (Ref. 13); however, the design of fuel transfer systems is such that pumps will operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day [and engine mounted] tanks during or following DG testing. In such a case, a 31 day Frequency is appropriate. Since proper operation of fuel transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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SR 3.8.1.7

See SR 3.8.1.2.

[SR 3.8.1.8

, (i.e., unit auxiliary transformer)

Transfer of each [4.16 kV ESF bus] power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads.
[The [18 month] Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

~~This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR .]~~

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. ~~[~~For this unit, the single load for each DG and its horsepower rating is ~~as follows:]~~. This Surveillance may be accomplished by:

an intake cooling water pump with a nameplate rating of 600 hp

St. Lucie – Unit 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power or while solely supplying the bus, or

b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

As required by IEEE-308 (Ref. 14), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The ~~[3]~~ seconds specified is equal to 60% of a ~~typical 5~~ second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the ~~system~~ must recover following load rejection~~. [The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 10).~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

This SR is modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns)

BASES

SURVEILLANCE REQUIREMENTS (continued)

provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR. Note 2 ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ ~~[~~0.9~~]~~. This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, Note 2 allows the Surveillance to be conducted at a power factor other than ≤ ~~[~~0.9~~]~~. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ ~~[~~0.9~~]~~ results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to ~~[~~0.9~~]~~ while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of ~~[~~0.9~~]~~ may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close as practicable to ~~[~~0.9~~]~~ without exceeding the DG excitation limits. 2

~~REVIEWER'S NOTE~~

~~The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable~~:

~~a. Performance of the SR will not render any safety system or component inoperable,~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and~~

~~c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

19

SR 3.8.1.~~10~~

6

Move Bases discussion to after SR 3.8.1.18

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG will not trip upon loss of the load. These acceptance criteria provide DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

~~[The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 10) and is intended to be consistent with expected fuel cycle lengths.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

4

~~]~~

2

This SR is modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbation to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system

BASES

SURVEILLANCE REQUIREMENTS (continued)

when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR. Note 2 ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ ~~[~~0.9~~]~~. This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, Note 2 allows the Surveillance to be conducted at a power factor other than ≤ ~~[~~0.9~~]~~. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ ~~[~~0.9~~]~~ results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to ~~[~~0.9~~]~~ while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of ~~[~~0.9~~]~~ may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close as practicable to ~~[~~0.9~~]~~ without exceeding the DG excitation limits. 2

~~----------REVIEWER'S NOTE----------~~
~~The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

~~a. Performance of the SR will not render any safety system or component inoperable,~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and~~

~~c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.~~
~~----------~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.1~~1~~

As required by Regulatory Guide 1.108 (Ref. 1~~0~~), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG auto-start time of ~~[~~10~~]~~ seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, high pressure injection systems are not capable of being operated at full flow, or shutdown cooling (SDC) systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

~~[The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

~~[~~ SR 3.8.1.1~~2~~

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (~~[~~10~~]~~ seconds) from the ~~design basis~~ actuation signal ~~(LOCA signal)~~ and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.1~~2~~.d and SR 3.8.1.1~~2~~.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The requirement to verify the connection of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, high pressure injection systems are not capable of being operated at full flow, or SDC systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

~~[The Frequency of [18 months] takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~----------REVIEWER'S NOTE----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 2

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that during operation with the reactor ~~–~~critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing 5

BASES

SURVEILLANCE REQUIREMENTS (continued)

OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.~~]~~ 2

SR 3.8.1.1~~3~~ 2 6

This Surveillance demonstrates that DG noncritical protective functions (e.g., high jacket water temperature) are bypassed on a loss of voltage signal concurrent with an ESF actuation ~~test~~ signal. Noncritical automatic trips are all automatic trips except: 1

a. Engine overspeed; and 1

b. Generator differential current~~;~~

~~[c. Low lube oil pressure;~~

~~d. High crankcase pressure; and~~

~~e. Start failure relay.]~~ 2

The noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

~~[The [18 month] Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 4 2

The SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

~~---------- REVIEWER'S NOTE ----------~~
~~The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

~~a. Performance of the SR will not render any safety system or component inoperable,~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and~~ 4

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.~~

SR 3.8.1.1~~4~~

Regulatory Guide 1.108 (Ref. 1~~0~~), paragraph 2.a.(3), requires demonstration that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ~~≥ [~~2~~]~~ hours of which is at a load equivalent to 110% of the continuous duty rating) and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

the 2 hour rating (i.e.,

This SR meets the intent of this requirement to the extent practical by loading the DG to approximately 104% of the engine continuous duty rating not to exceed the 2000 hour rating. This loading requirement challenges the full loading capability of the DG without unnecessarily overloading the DG.

The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

~~[The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance is modified by three Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. ~~The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.~~ 2

Note 2 provides an allowance that c

~~C~~redit may be taken for unplanned events that satisfy this SR. Note 3 ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ ~~[~~0.9~~]~~. This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, Note 3 allows the Surveillance to be conducted at a power factor other than ≤ ~~[~~0.9~~]~~. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ ~~[~~0.9~~]~~ results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to ~~[~~0.9~~]~~ while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of ~~[~~0.9~~]~~ may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close practicable to ~~[~~0.9~~]~~ without exceeding the DG excitation limits. 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

Move Bases discussion to after SR 3.8.1.19

SR 3.8.1.~~15~~ (20) (6)

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within ~~[~~10~~]~~ seconds. The ~~[~~10~~]~~ second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. (2)
~~[The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(5).~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------~~ (4)
~~]~~ (2)

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least ~~[~~2~~]~~ hours at full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.1~~6~~ (4) (6)

As required by Regulatory Guide 1.108 (Ref. 1~~0~~), paragraph 2.a.(6), this (1) (1)
Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and that the DG can be returned to ready to load status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the DG to reload if a

BASES

SURVEILLANCE REQUIREMENTS (continued)

subsequent loss of offsite power occurs. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open and can receive and autoclose signal on bus undervoltage, and the load sequence timers are reset. 5

~~[The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 4
~~]~~ 2

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to ready to load operation if a LOCA actuation signal is received during operation in the test mode. Ready to load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 14), paragraph 6.2.6(2).

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

[The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(8); takes into consideration unit conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
----------]

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing

BASES

SURVEILLANCE REQUIREMENTS (continued)

following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR. ~~]~~ 2

6

SR 3.8.1.1~~8~~ 6

Under accident ~~[~~and loss of offsite power~~]~~ conditions loads are sequentially connected to the bus by the ~~[automatic~~ load sequence~~r]~~. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The ~~[10]%~~ load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. ~~Reference 1 provides a summary of the automatic loading of ESF buses.~~ 3

timers

±1 second

1

~~[The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(2); takes into consideration unit conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.~~

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

~~REVIEWER'S NOTE~~

~~The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

~~a. Performance of the SR will not render any safety system or component inoperable,~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and~~

~~c. Performance of the SR or failure of the SR will not cause or result in an AOO with attendant challenge to plant safety systems.~~

4

SR 3.8.1.1~~9~~

7

6

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.~~1~~, during a loss of offsite power actuation ~~test~~ signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

~~[The Frequency of [18 months] takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of [18 months].~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they

BASES

SURVEILLANCE REQUIREMENTS (continued)

are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.~~20~~ 18 6

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.

~~[The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 10).~~

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 4 2

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated, and temperature maintained consistent with manufacturer recommendations.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter ~~[~~8~~]~~. (U)
3. Regulatory Guide 1.9, Rev. ~~[~~3~~]~~. (0)
4. FSAR, Chapter ~~[~~6~~]~~. (U)

1 2 1

BASES

REFERENCES (continued)

5. FSAR, Chapter [15].

6. Regulatory Guide 1.93, Rev. [ ], [date].

7. NRC Letter, "St. Lucie Unit No. 2 – Issuance of Amendment Regarding Diesel Generator Allowed Outage Time Extension (TAC No. MA7206)," dated April 26, 2001 (ML011210391).

~~7~~. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

~~8~~. Generic Letter 84-15.

~~9~~. 10 CFR 50, Appendix A, GDC 18.

~~10~~. Regulatory Guide 1.108, Rev. [1], [August 1977].

~~11~~. Regulatory Guide 1.137, Rev. [ ], [date].

~~12~~. ANSI C84.1-198~~2~~.

~~13~~. ~~ASME Code for Operation and Maintenance of Nuclear Power Plants.~~

14. IEEE Standard 308-[197~~8~~].

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.1 BASES, AC SOURCES - OPERATING

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Discussions regarding load sequencers have been deleted, because PSL does not use load sequencers. Each load is sequenced with the use of its associated load sequence timer.

4. The Reviewer's Note has been deleted, because it is not meant to be retained in the plant specific ITS submittal.

5. Editorial/grammatical error corrected.

6. Changes have been made to be consistent with changes made to the Specifications.

7. Corrections have been made consistent with the Writer’s Guide for the Improved Technical Specifications, TSTF-GG-05-01, Section 3.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.8.1, AC SOURCES - OPERATING

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

ITS 3.8.2, AC Sources - Shutdown

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ELECTRICAL POWER SYSTEMS

AC SOURCES -

SHUTDOWN

LIMITING CONDITION FOR OPERATION

T

LCO 3.8.2 3.8.1.2 ~~As a minimum, t~~he following A~~.~~C~~.~~ electrical power sources shall be OPERABLE:

qualified

LCO 3.8.2.a a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

AC electrical power

A02

sub (s) required by LCO 3.8.10, "Distribution Systems Shutdown"

LCO 3.8.2.b b. One diesel generator set ~~with~~:

capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

A03

1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

See ITS 3.8.1

2. A fuel storage system containing a minimum of 19,000 gallons of fuel, and

See ITS 3.8.3

3. A fuel transfer pump.

See ITS 3.8.1

Applicability **APPLICABILITY**: MODES 5 and 6~~.~~,

During movement of irradiated fuel assemblies.

M01

ACTION:

Insert proposed ACTIONS NOTE

NOTE

applicable Conditions and Required Actions of LCO 3.8.10,

ACTION A NOTE Enter ~~the ACTION of LCO 3.8.2.2, "A.C. Distribution - Shutdown,"~~ with one required train de-energized as a result of ~~inoperable offsite circuit~~.

Condition A

Add proposed Required Action A.1 and Completion Time

assemblies

L01

ACTION A ACTION B With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations ~~involving CORE ALTERATIONS, operations~~ involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, movement of irradiated fuel~~, or crane operation with loads over the fuel storage pool~~. In addition, ~~when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the top of irradiated fuel assemblies seated within the reactor vessel,~~ immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

ACTION A.2.1, A.2.2, B.1, and B.2

L02

ACTION A.2.3 and B.3

M02

SURVEILLANCE REQUIREMENTS

Insert proposed SR 3.8.2.1 NOTE

L04

SR 3.8.2.1 4.8.1.2.1 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of ~~4.8.1.1.1 and 4.8.1.1.2~~ ~~except for requirement 4.8.1.1.2a.5~~.

Specification 3.8.1 Surveillances

L05

Add proposed SR 3.8.2.1 exceptions

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO 3.8.2 3.8.1.2 ~~As a minimum, t~~T~~he~~ he following A~~.~~C~~.~~ electrical power sources shall be OPERABLE:

LCO 3.8.2.a a. One ~~circuit~~ qualified circuit between the offsite transmission network and the onsite Class 1E distribution system, and

A02

AC electrical power sub (s) required by LCO 3.8.10, "Distribution Systems Shutdown"

LCO 3.8.2.b b. One diesel generator ~~with:~~ capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

A03

1. Two engine-mounted fuel tanks containing a minimum volume of 238 gallons of fuel,

See ITS 3.8.1

2. A fuel storage system containing a minimum volume of 42,500 gallons of fuel, and

3. A fuel transfer pump.

See ITS 3.8.1

See ITS 3.8.3

Applicability **APPLICABILITY:** MODES 5 and 6~~.~~, During movement of recently irradiated fuel assemblies.

M01

ACTION:

Insert proposed ACTIONS NOTE

NOTE

applicable Conditions and Required Actions of LCO 3.8.10,

ACTION A NOTE Enter ~~the ACTION of LCO 3.8.3.2, "Onsite Power Distribution - Shutdown,"~~ with one required train de-energized as a result of ~~inoperable offsite circuit~~ Condition A.

Add proposed Required Action A.1 and Completion Time

ACTION A ACTION B With less than the above minimum required A.C. electrical power sources OPERABLE, assemblies immediately suspend all operations ~~involving CORE ALTERATIONS, operations~~ involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, movement of irradiated fuel~~, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to 3.58 square inch vent~~. In addition, ~~when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange,~~ immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible~~.~~

L01

ACTION A.2.1, A.2.2, B.1, and B.2

L02

L03

M02

ACTION A.2.3 and B.3

SURVEILLANCE REQUIREMENTS

Insert proposed SR 3.8.2.1 NOTE

L04

SR 3.8.2.1 4.8.1.2.1 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of ~~4.8.1.1.1 and 4.8.1.1.2 (except for requirement 4.8.1.1.2a.5)~~.

Specification 3.8.1 Surveillances

Add proposed SR 3.8.2.1 exceptions

L05

DISCUSSION OF CHANGES
ITS 3.8.2, AC SOURCES - SHUTDOWN

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.8.1.2.a states one circuit between the offsite transmission network and the onsite Class 1E distribution system shall be OPERABLE. ITS 3.8.2.a states one qualified circuit between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System shall be OPERABLE. This changes the CTS requirement by modifying the description of the offsite circuit to the ITS requirement for the circuit description of “qualified.”

The change is acceptable because the ITS requirement for the offsite circuit remains consistent with the intent of the CTS requirement. The ITS requirement of being “qualified” provides a clarification for the offsite circuit requirements consistent with the existing design requirements for these circuits. The change in wording does not modify the technical requirement for the CTS. The ITS requirement is consistent with the ISTS wording for this requirement. This change is designated as administrative because the technical requirements of the specifications have not changed.

A03 CTS 3.8.1.2.b requires one diesel generator set to be OPERABLE. ITS LCO 3.8.2 b states “One diesel generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10,” shall be OPERABLE. This changes the CTS by stating that the DG set is capable of supplying the required electrical power to the distribution subsystem(s) required by LCO 3.8.10.

This change is acceptable because the ITS requirement for the DG set remains consistent with the intent of the less explicit CTS requirement. The addition of the ISTS wording, “capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10,” provides a clarifying statement for the CTS requirements that is consistent with the DG OPERABILITY requirements in these MODES. The ITS requirement is consistent with the ISTS wording for this requirement. This change is designated as administrative because the technical requirements of the specifications have not changed.

MORE RESTRICTIVE CHANGES

M01 CTS 3.8.1.2 is applicable during MODES 5 and 6. ITS 3.8.2 is applicable in MODES 5 and 6, and during the movement of irradiated fuel assemblies. In addition, a Note has been added to the ACTIONS of ITS 3.8.2 that states

LCO 3.0.3 is not applicable. This changes the CTS by requiring the AC Sources to be OPERABLE under more conditions than is currently required.

The purpose of CTS 3.8.1.2 is to ensure that sufficient AC Sources are available to mitigate the consequences of an analyzed event during shutdown modes. This change also provides an explicit requirement that the AC Sources must be OPERABLE during the movement of irradiated fuel assemblies. The movement of irradiated fuel assemblies may occur during MODE 5 or 6, however movement of irradiated fuel assemblies could also occur while the unit is operating or defueled if moving fuel only in the spent fuel pool. CTS 3.8.1.1 (ITS 3.8.1) does not provide the appropriate compensatory actions under this condition. The activity should be suspended immediately when the AC Sources are not available consistent with the immediate actions for CORE ALTERATIONS in the CTS 3.8.1.2 Action. Because the actions provided in LCO 3.0.3 will not place the unit in a safe condition with respect to the applicable condition, a modifying Note is added to the ITS ACTIONS. This change is acceptable because the proposed Applicability is consistent with the Applicability in the AC Distribution System – Shutdown Specification (ITS 3.8.10). AC Sources provide the power for the AC Electrical Distribution System. This change is designated as more restrictive because the Applicability of the Specification, although modified, has been expanded.

M02 The Unit 1 CTS 3.8.1.2 Action requires, in part, immediate initiation of corrective action to restore the sources to OPERABLE status with less than the minimum AC electrical power sources in MODE 5 with reactor coolant loops not filled or in MODE 6 with water level less than 23 feet above the top of irradiated fuel assemblies seated within the reactor vessel. Unit 2 CTS 3.8.1.2 Action requires, in part, immediate initiation of corrective action to restore the sources to OPERABLE status with less than the minimum AC electrical power sources in MODE 5 with reactor coolant loops not filled or in MODE 6 with water level less than 23 feet above the reactor vessel flange. ITS 3.8.2 Required Actions A.2.3 and B.3 require the immediate initiation of action to restore the required AC Sources to OPERABLE status. This changes the CTS by expanding the Action requirement to restore the inoperable AC Sources to OPERABLE status regardless of fill status of the reactor coolant loops or the water level above the top of irradiated fuel (Unit 1) or reactor vessel flange (Unit 2).

The purpose of ITS 3.8.1.2 Required Actions A.2.3 and B.3 is to promptly initiate action to restore the LCO requirements. When a required offsite circuit or one required DG is inoperable, the actions imposed by the CTS 3.8.1.2 ACTION do not necessarily place the unit in a MODE or other specified condition in which CTS 3.8.1.2 is not applicable. Therefore, proposed ITS 3.8.2 Required Actions A.2.3 and B.3 are being expanded to apply in all plant configurations in MODES 5 and 6. When in MODE 5 with the reactor coolant loops filled and MODE 6 with water level ≥ 23 ft above the reactor vessel flange, at least one shutdown cooling loop is required to be OPERABLE and in operation and two trains of the Control Room Emergency Ventilation System are required to be OPERABLE during movement of irradiated fuel assemblies. Therefore, AC Sources are, at a minimum, required to be OPERABLE to support these required systems to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power. These additional restrictions will ensure

action is immediately taken to restore compliance with the LCO requirements. This change is designated as more restrictive because the CTS Required Actions are expanded apply to all plant configurations in MODES 5 and 6 and during movement of irradiated fuel assemblies.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* The CTS 3.8.1.2 ACTION requires the suspension of certain activities when the minimum required AC electrical power sources are inoperable. ITS 3.8.2 provides an alternate Required Action (ITS 3.8.2 Required Action A.1) that allows the declaration of affected required feature(s) with no offsite power available inoperable instead of requiring the specified activities to be suspended. This changes the CTS by allowing the affected required feature(s) with no offsite power available to be declared inoperable instead of suspending the specified activities.

The purpose of CTS 3.8.1.2 is to ensure the appropriate offsite circuit is OPERABLE. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a loss of offsite power occurring during the repair period. This changes the CTS by allowing the affected required feature(s) with no offsite power available to be declared inoperable instead of suspending specified activities (i.e., movement of irradiated fuel assemblies). Since the ITS 3.8.2 circuit OPERABILITY requirements are proposed to require supplying power to all required electrical power distribution subsystems, if one or more subsystems are not powered by an offsite circuit, that circuit is inoperable. Conservative actions can be assured if required equipment with no offsite power is declared inoperable and the associated ACTIONS of the individual equipment taken (ITS 3.8.2 Required Action A.1). This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.8.1.2 ACTION states, in part, with less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS …or crane operation with loads over the fuel storage pool. ITS 3.8.2 ACTIONS do not include these actions. This changes the CTS Actions by

removing actions to immediately suspend all operations involving CORE ALTERATIONS or crane operation with loads over the fuel storage pool.

The purpose of the CTS 3.8.1.2 ACTION is to minimize the possibility of an event that may require the AC source to mitigate the consequences of a design basis event. CORE ALTERATIONS is defined in CTS 1.9, in part, as "the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity, within the reactor vessel with the head removed and fuel in the vessel." CORE ALTERATIONS only occur when the reactor vessel head is removed - it only applies in MODE 6. There is only one accident considered during MODE 6 that involves a CORE ALTERATION: a fuel handling accident. According to the Unit 1 UFSAR Section 15.4.3 and Unit 2 UFSAR Section 15.7.4.1.2, a fuel handling accident is initiated by the dropping of an irradiated fuel assembly either in the containment or in the fuel building. Suspension of CORE ALTERATIONS, except for suspension of movement of irradiated fuel, will not prevent or impair the mitigation of a fuel handling accident.

Crane travel related requirements were relocated from the CTS in Unit 1 and Unit 2 PSL Amendments 190 and 134, respectively, ("St. Lucie Units 1 and 2 – Issuance of Amendments Regarding the Relocation of Spent Fuel Pool Crane Technical Specification Requirements (TAC NOS. MB5667 and MB5668)," dated April 28, 2004 (ADAMS Accession No. ML 040440111)) to the respective unit Updated Final Safety Analysis Report where the operational detail is controlled under 10 CFR 50.59 criteria. As cited in the NRC Safety Evaluation associated with Amendments 190 (Unit 1) and 134 (Unit 2), NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," regulatory guidelines for control of heavy load lifts provide assurance of safe handling of heavy loads in areas where a load drop could impact stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. Section 5.1.1 of NUREG-0612 provides guidelines for reducing the likelihood of dropping heavy loads and provides criteria for establishing safe load paths; procedures for load-handling operations; training of crane operators; design, testing, inspection, and maintenance of cranes and lifting devices; and analyses of the impact of heavy load drops. The guidelines in Sections 5.1.2 through 5.1.6 address alternatives to either further reduce the probability of a load-handling accident or mitigate the consequences of heavy load drops. These alternatives include using a single-failure-proof crane to improve reliability through increased factors of safety and through redundancy or duality in certain active components. Criteria for design of single-failure-proof cranes are included in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants."

This change is acceptable because the possibility of damage to a fuel assembly as a consequence of mishandling components other than an irradiated fuel assembly is minimized by thorough training, detailed procedures and equipment design. The PSL crane design precludes the handling of heavy objects, such as shipping casks, over the spent fuel pool storage racks. Administrative controls prevent the movement of heavy loads over the cask pit whenever the cask pit rack is installed in the cask area of the spent fuel storage pool. In addition, the cask handling crane design meets the regulatory guidance for single-failure-proof cranes in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants" and NUREG-0612. Administrative controls that control of movement of light

loads or prevent movement of light loads over irradiated fuel assemblies are similar to those used for control of heavy loads, to the extent practicable, as advised in NUREG-0612. Consequently, the possibility of dropping a load other than an irradiated fuel assembly and damaging of fuel assemblies in the spent fuel storage pool is remote. Therefore, the CTS 3.8.1.2 action related to suspension of crane operation with loads over the fuel storage pool is not necessary to be included in the technical specifications and are removed.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 **Unit 2 Only** *(Category 4 – Relaxation of Required Action)* CTS 3.8.1.2 ACTION requires, in part, with less than the minimum required AC electrical power sources OPERABLE, the Reactor Coolant System (RCS) to be depressurized and vented through a greater than or equal to 3.58 square inch vent within 8 hours. ITS 3.8.2 does not include this required action.

The purpose of CTS is to ensure the minimum AC electrical power sources are OPERABLE. CTS 3.4.9.3 provides specific OPERABILITY requirements for overpressure protection systems when in MODE 5 and MODE 6 with the reactor vessel head on. The definition of OPERABLE in CTS 1.19 requires necessary electrical power to components so they can perform their required safety function. This change is acceptable because CTS 3.4.9.3 Action a.1 requires depressurization and venting of the RCS when adequate pressure relief methods are inoperable, including the loss of electrical power necessary to support the required pressure relief components. ITS 3.4.12 retains the requirement to depressurize and establish the required RCS vent size (Refer to ITS 3.4.12 for any changes to CTS 3.4.9.3 requirements). Therefore, the CTS 3.8.1.2 action to depressurize the RCS and vent through a greater than or equal to 3.58 square inch vent is unnecessary. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L04 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.8.1.2.1 requires the AC electrical power sources to be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.5. ITS SR 3.8.2.1 includes the performance exception in the Note to SR 3.8.2.1. However, the Note provides additional ITS 3.8.1 SRs exempt from being required to be performed. ITS SR 3.8.2.1 Note states the following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.9, SR 3.8.1.10, SR 3.8.1.12 through SR 3.8.1.14, and SR 3.8.1.16. The Note to Unit 2 ITS SR 3.8.2.1 also includes SR 3.8.1.19 and SR 3.8.1.20.

This changes the performance Frequency of specific CTS 4.8.1.1 Surveillances required by CTS 4.8.1.2.1 by suspending the performance requirement of the Surveillances in MODES 5 and 6 and other specified conditions in the Applicability until the AC Source is no longer required to meet the AC Sources – Shutdown LCO.

The purpose of CTS 3.8.1.2 is to ensure the AC Sources required when in MODE 5 or 6 or during movement of irradiated fuel assemblies are demonstrated

to be OPERABLE. Currently CTS 4.8.1.1.2.a.5 is not required to be performed (however it must be met). CTS 4.8.1.1.2.e.2 (ITS SR 3.8.1.9) demonstrates the DG capability to reject the single largest load, Unit 1 CTS 4.8.1.1.2.e.3 and Unit 2 CTS 4.8.1.1.2.e.4 (ITS SR 3.8.1.10) verifies a DG start on a loss of offsite power, Unit 2 CTS 4.8.1.1.2.e.3 (ITS SR 3.8.1.19) verifies that the DG does not trip on a load reject, Unit 1 CTS 4.8.1.1.2.e.5.c and Unit 2 CTS 4.8.1.1.2.e.6.c (ITS 3.8.1.12) demonstrates the DG noncritical protective functions are bypassed on a loss of voltage signal concurrent with an ESF actuation test signal, Unit 1 CTS 4.8.1.1.2.e.6 and Unit 2 CTS 4.8.1.1.2.e.7 (ITS SR 3.8.1.13) is the DG 24 hour run test, Unit 2 CTS 4.8.1.1.2.e.13 (ITS SR 3.8.1.20) is the DG hot restart test, Unit 1 CTS 4.8.1.1.2.e.8 and Unit 2 CTS 4.8.1.1.2.e.9 (ITS SR 3.8.1.14) ensures manual synchronization and automatic load transfer from the DG to the offsite source can be made and the DG can be returned to ready to load status when offsite power is restored, and Unit 1 CTS 4.8.1.1.2.e.11 and Unit 2 CTS 4.8.1.1.2.e.12 (ITS SR 3.8.1.16) is a test of the individual load sequence timers.

The tests listed in the proposed SR 3.8.2.1 Note normally require the DG to be paralleled with offsite power. With limited AC Sources available, a single event could compromise both the required offsite circuit and the DG, which presents a significant risk of a station blackout. The NRC has previously recognized this in the exception of CTS 4.8.1.1.2.a.5 in CTS 4.8.1.2.1. In an effort to consistently address this concern and to avoid potential conflicting Technical Specifications, the Surveillances that would require a DG to be connected to the offsite source or would require disconnection of the required offsite circuit and de-energization of required buses are excepted from being performed. The exception does not remove the requirement for the required DG to be capable of performing its related support function. The exception only removes the requirement to demonstrate the capability while that source of power is being relied on to meet the supporting LCO.

This change is acceptable because the SR specified in the Note will continue to be required to be capable of being met. Historically, the most probable result of performing a Surveillance is the verification of conformance with the requirements while failure to meet a Surveillance is typically discovered when not performing a surveillance test. Therefore, any risk of suspending Surveillance performance until the AC Source is no longer required to meet the AC Sources Shutdown LCO is offset by minimizing the risk of losing both required AC sources. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L05 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.8.1.2.1 requires the AC electrical power sources to be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of CTS 4.8.1.1.1 and CTS 4.8.1.1.2 except for CTS 4.8.1.1.2.a.5. ITS SR 3.8.2.1 has included a similar allowance to waive performance of the current Surveillance (ITS SR 3.8.1.3) in the Note to SR 3.8.2.1. However, ITS SR 3.8.2.1 exempts SRs from being required. ITS SR 3.8.2.1 does not require the following ITS 3.8.1 SRs: SR 3.8.1.8, SR 3.8.1.11, SR 3.8.1.15, SR 3.8.17, and SR 3.8.1.18.

This changes the CTS by not requiring specific CTS 4.8.1.1 Surveillances in MODES 5 and 6 and other specified conditions in the Applicability.

DISCUSSION OF CHANGES
ITS 3.8.2, AC SOURCES - SHUTDOWN

The purpose of CTS 3.8.1.2 is to ensure the AC Sources required when in MODE 5 or 6 or during movement of irradiated fuel assemblies are demonstrated OPERABLE. This change is acceptable because the deleted Surveillance Requirements are not necessary to verify that the required AC Sources can perform their required support function. Thus, the required AC Sources continue to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed electrical support function.

This change deletes the following CTS 4.8.1.1 Surveillances from being required in MODE 5 or 6 or during movement of irradiated fuel assemblies: CTS 4.8.1.1.1.b (ITS SR 3.8.1.8), the automatic and manual transfer of power sources from normal circuit (unit auxiliary transformer) to the offsite circuit (startup transformer), Unit 1 CTS 4.8.1.1.2.e.4 and Unit 2 CTS 4.8.1.1.2.e.5 (ITS SR 3.8.1.11), the ESF actuation signal DG start test, Unit 1 CTS 4.8.1.1.2.e.9 and Unit 2 CTS 4.8.1.1.2.e.10 (ITS SR 3.8.1.15), ESF signal override with DG operating in test mode test, Unit 1 CTS 4.8.1.1.2.e.5 and Unit 2 CTS 4.8.1.1.2.e.6 (ITS SR 3.8.1.17), ESF concurrent with loss of offsite power signal test, and CTS 4.8.1.1.2.f (ITS SR 3.8.1.18), the simultaneous DG start test.

ITS SR 3.8.1.8 is not required to be met since only one offsite circuit is required by ITS LCO 3.8.2. ITS SR 3.8.1.11 and SR 3.8.1.17 are not required to be met because the ESF actuation signal is not required to be OPERABLE in MODE 5 or 6 or during movement of irradiated fuel assemblies. A single event could compromise both the required offsite circuit and the DG, which presents a significant risk of a station blackout. Therefore, SR 3.8.1.15 is not required to be met so the DG is not required to synchronize to the offsite circuit. SR 3.8.1.18 is excepted because starting independence is not required for the DG that is not required to be OPERABLE. The required DG is required to start if a loss of offsite power occurs. Thus, the requirement to autostart the required DG on a loss of offsite power signal (ITS SR 3.8.1.10) continues to be required by ITS SR 3.8.2.1. This change is designated as less restrictive because Surveillances that are required in CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

3.8.1.2 LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

3.8.1.2.a a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems Shutdown" and

3.8.1.2.b b. One diesel generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

Applicability DOC M01 APPLICABILITY: MODES 5 and 6,
During movement of ~~[recently]~~ irradiated fuel assemblies. (2)

ACTIONS

----------NOTE----------
DOC M01 LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action A. One required offsite circuit inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. --- DOC L01 A.1 Declare affected required feature(s) with no offsite power available inoperable. OR	Immediately

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.1 Suspend movement of [recently] irradiated fuel assemblies.	Immediately
	AND	
	A.2.2 Suspend operations involving positive reactivity additions that could result in a loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.3 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. One required DG inoperable.	B.1 Suspend movement of [recently] irradiated fuel assemblies.	Immediately
	AND	
	B.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	B.3 Initiate action to restore required DG to OPERABLE status.	Immediately

Action DOC L02 2

Action

Action DOC M02

Action 2

Action

Action DOC M02

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 4.8.1.2.1 DOC L04	SR 3.8.2.1	----------NOTE---------- The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.9 ~~through~~ SR 3.8.1.~~11~~, SR 3.8.1.~~13~~ through SR 3.8.1.~~16~~, and [SR 3.8.1.~~18~~]. ----------
DOC L05		For AC sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources - Operating," except SR 3.8.1.8, SR 3.8.1.~~12~~, SR 3.8.1.~~17~~, SR 3.8. 1.~~19~~, and SR 3.8.1.~~20~~, are applicable.	In accordance with applicable SRs

10 12 14 16 11 15 17 18

3 2 3

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

3.8.1.2 LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

3.8.1.2.a a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems Shutdown" and

3.8.1.2.b b. One diesel generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

Applicability DOC M01 APPLICABILITY: MODES 5 and 6,
During movement of ~~[recently]~~ irradiated fuel assemblies. (2)

ACTIONS

----------NOTE----------
DOC M01 LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action A. One required offsite circuit inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. --- DOC L01 A.1 Declare affected required feature(s) with no offsite power available inoperable. OR	Immediately

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately
	AND	
	A.2.2 Suspend operations involving positive reactivity additions that could result in a loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.3 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. One required DG inoperable.	B.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately
	AND	
	B.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	B.3 Initiate action to restore required DG to OPERABLE status.	Immediately

Action
DOC L02
2
Action
DOC M02
Action
Action
DOC L02
2
Action
DOC M02
Action

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
SR 4.8.1.2.1 DOC L04	SR 3.8.2.1	----------NOTE---------- The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.9 ~~through~~ SR 3.8.1.~~11~~ 10, SR 3.8.1.~~13~~ 12 through SR 3.8.1.~~16~~ 14, ~~and~~ ~~[SR 3.8.1.18]~~ SR 3.8.1.16, SR3.8.1.19, and SR 3.8.1.20. ----------	3 2
DOC L05		For AC sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources - Operating," except SR 3.8.1.8, SR 3.8.1.~~12~~ 11, SR 3.8.1.~~17~~ 15, SR 3.8.1.~~19~~ 17, and SR 3.8.1.~~20~~ 18, are applicable.	In accordance with applicable SRs 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.2, AC SOURCES - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The SRs have been changed to be consistent with the changes made to the Surveillance Requirements in ITS 3.8.1.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of ~~[recently]~~ irradiated fuel assemblies ensures that: (2)

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident ~~[involving handling recently irradiated fuel~~. ~~Due to radioactive decay, AC electrical power is only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~. (2)

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a

BASES

APPLICABLE SAFETY ANALYSES (continued)

significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are ~~generally~~ planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on: 1

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents ~~[involving handling recently irradiated fuel]~~). 2

The qualified offsite circuit must be capable of providing three phases of AC power, maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es). Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit. 1

U

BASES

LCO (continued)

~~[Offsite circuit #1 consists of Safeguards Transformer B, which is supplied from Switchyard Bus B, and is fed through breaker 52-3 powering the ESF transformer XNBO1, which, in turn, powers the #1 ESF bus through its normal feeder breaker. The second offsite circuit consists of the Startup Transformer, which is normally fed from the Switchyard Bus A, and is fed through breaker PA 0201 powering the ESF transformer, which, in turn, powers the #2 ESF bus through its normal feeder breaker.]~~ 2

Insert 1

degraded or

The DG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective ESF bus on detection of bus undervoltage, and accepting required loads. This sequence must be accomplished within ~~[~~10~~]~~ seconds. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby at ambient conditions. 1 2

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

~~[In addition, proper sequencer operation is an integral part of offsite circuit OPERABILITY since its inoperability impacts on the ability to start and maintain energized loads required OPERABLE by LCO 3.8.10.]~~ 3

~~It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains.~~ 1

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of ~~[recently]~~ irradiated fuel assemblies provide assurance that: 2

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies,

b. Systems needed to mitigate a fuel handling accident ~~[involving handling [recently] irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~ are available, 2

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

2

Insert 1

An offsite circuit consists of a unit startup transformer and the associated circuit path, including non-ESF buses, up to and including the feeder breaker to the respective unit 4.16 kV ESF bus. Alternately, when the unit startup transformer is unavailable, the associated offsite circuit may be supplied from the opposite unit startup transformer via the associated non-ESF buses. The Unit 2 startup transformer may only be utilized as a qualified offsite AC source to Unit 1 provided it is not supplying power to a Unit 2 bus while operating in MODE 1, 2, 3, or 4. The Unit 2 startup transformer may supply power to ESF buses of both units when both units are in MODE 5 or 6 or defueled. | R1

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF train. Although two trains may be required by LCO 3.8.10, the remaining train with offsite power available may be capable of supporting sufficient required features to allow continuation of ~~[recently]~~ irradiated fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS. (2)

A.2.1, A.2.2, A.2.3, B.1, B.2, and B.3

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend movement of ~~[recently]~~ irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron (2)

BASES

ACTIONS (continued)

concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive ~~MTC~~ must also be evaluated to ensure they do not result in a loss of required SDM.

moderator temperature coefficient

4

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS are not entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 provides the appropriate restrictions for the situation involving a de-energized train.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.~~12~~ and SR 3.8.1.~~19~~ are not required to be met because the ESF actuation signal is not required to be OPERABLE. SR 3.8.1.~~17~~ is not required to be met because the required OPERABLE DG~~(s)~~ is not required to undergo periods of being synchronized to the offsite circuit. SR 3.8.1.~~20~~ is excepted because starting independence is not required ~~with DG(s) that are~~ not required to be OPERABLE.

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for the DG that is

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG~~(s)~~ from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 4160 V ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC Sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR. 1

REFERENCES None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of ~~[recently]~~ irradiated fuel assemblies ensures that: (2)

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident ~~[involving handling recently irradiated fuel. Due to radioactive decay, AC electrical power is only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~. (2)

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a

BASES

APPLICABLE SAFETY ANALYSES (continued)

significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are ~~generally~~ planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on: 1

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents ~~[involving handling recently irradiated fuel]~~). 2

The qualified offsite circuit must be capable of providing three phases of AC power, maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es). Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit. 1

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BASES

LCO (continued)

~~[Offsite circuit #1 consists of Safeguards Transformer B, which is supplied from Switchyard Bus B, and is fed through breaker 52-3 powering the ESF transformer XNBO1, which, in turn, powers the #1 ESF bus through its normal feeder breaker. The second offsite circuit consists of the Startup Transformer, which is normally fed from the Switchyard Bus A, and is fed through breaker PA 0201 powering the ESF transformer, which, in turn, powers the #2 ESF bus through its normal feeder breaker.]~~ 2

Insert 1

degraded or

The DG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective ESF bus on detection of bus undervoltage, and accepting required loads. This sequence must be accomplished within ~~[~~10~~]~~ seconds. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby at ambient conditions. 1 2

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

~~[In addition, proper sequencer operation is an integral part of offsite circuit OPERABILITY since its inoperability impacts on the ability to start and maintain energized loads required OPERABLE by LCO 3.8.10.]~~ 3

~~It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains.~~ 1

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of ~~[recently]~~ irradiated fuel assemblies provide assurance that: 2

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies,

b. Systems needed to mitigate a fuel handling accident ~~[involving handling [recently] irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~ are available, 2

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

2

Insert 1

An offsite circuit consists of a unit startup transformer and the associated circuit path, including non-ESF buses, up to and including the feeder breaker to the respective unit 4.16 kV ESF bus. Alternately, when the unit startup transformer is unavailable, the associated offsite circuit may be supplied from the opposite unit startup transformer via the associated non-ESF buses. The Unit 1 startup transformer may only be utilized as a qualified offsite AC source to Unit 2 provided it is not supplying power to a Unit 1 bus while operating in MODE 1, 2, 3, or 4. The Unit 1 startup transformer may supply power to ESF buses of both units when both units are in MODE 5 or 6 or defueled. |R1

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF train. Although two trains may be required by LCO 3.8.10, the remaining train with offsite power available may be capable of supporting sufficient required features to allow continuation of ~~[recently]~~ irradiated fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS. (2)

A.2.1, A.2.2, A.2.3, B.1, B.2, and B.3

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend movement of ~~[recently]~~ irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron (2)

BASES

ACTIONS (continued)

concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive ~~MTC~~ must also be evaluated to ensure they do not result in a loss of required SDM.

moderator temperature coefficient

4

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS are not entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 provides the appropriate restrictions for the situation involving a de-energized train.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.~~12~~ and SR 3.8.1.~~19~~ are not required to be met because the ESF actuation signal is not required to be OPERABLE. SR 3.8.1.~~17~~ is not required to be met because the required OPERABLE DG~~(s)~~ is not required to undergo periods of being synchronized to the offsite circuit. SR 3.8.1.~~20~~ is excepted because starting independence is not required ~~with DG(s) that are~~ not required to be OPERABLE.

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for the DG that is

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG~~(s)~~ from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 4160 V ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC Sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR. 1

REFERENCES None.

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.2 BASES, AC SOURCES - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Discussions regarding automatic load sequencers have been deleted, because PSL does not use load sequencers. Each electrical load is sequenced by its associated load sequence timers.

4. Corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01, Section 3.2.2.a.

5. Changes have been made to be consistent with changes made to the Specifications.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.8.2, AC SOURCES - SHUTDOWN

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

ITS 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

See ITS 3.8.1

Add proposed LCO 3.8.3

A02

LCO 3.8.3 3.8.1.1 ~~As a minimum, the following A.C. electrical power sources shall be OPERABLE:~~

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generator sets each with:

1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

SR 3.8.3.1

2. A separate fuel storage system containing a minimum of ~~19,000 gallons~~ of fuel, and

LA01

a 7 day supply

M02

A02

3. A separate fuel transfer pump.

See ITS 3.8.1

Applicability **APPLICABILITY**: ~~MODES 1, 2, 3 and 4.~~

When associated DG is required to be OPERABLE

A03

ACTIONS **ACTION**:

Add proposed ACTIONS Note

Add proposed ACTION A

Add proposed ACTIONS C and D

L01

L02

a. With one offsite circuit of 3.8.1.1.a inoperable, except as provided in Action f below:

1. Demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter.

2. Within 24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required features(s), declare required features(s) with no offsite power available inoperable when its redundant required features(s) is inoperable.

3. Restore the offsite circuit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

4. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.8.1

Add proposed ACTIONS B and E

M01

M01

L01

Add proposed ACTION F

L02

ELECTRICAL POWER SYSTEMS

ACTION (continued)

See ITS 3.8.1

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in the at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN. Following restoration of one diesel generator unit, follow ACTION Statement b. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator.

f. With one Unit 1 startup transformer (1A or 1B) inoperable and with a Unit 2 startup transformer (2A or 2B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 2 require the use of the startup transformer administratively available to both units, Unit 1 shall demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

g. LCO 3.0.4.b is not applicable to diesel generators.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments, indicated power availability; and

b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by transferring (manually and automatically) unit power supply from the auxiliary transformer to the startup transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

SR 3.8.3.1 a. In accordance with the Surveillance Frequency Control Program by:

See ITS 3.8.1

1. Verifying fuel level in the engine-mounted fuel tank,

SR 3.8.3.1 2. Verifying the fuel level in the fuel storage tank,

See ITS 3.8.1

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the engine-mounted tank,

Add proposed SR 3.8.3.2

M01

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

See ITS 3.8.1

4. Verifying the diesel starts from ambient condition and accelerates to approximately 900 rpm in less than or equal to 10 seconds**. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal**. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual/Local

   b) Simulated loss-of-offsite power by itself.

   c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.

   d) An ESF actuation test signal by itself.

5. Verifying the generator is synchronized, loaded to greater than or equal to 3500 kW in accordance with the manufacturer's recommendations and operates within a load band of 3300 to 3500 kW*** for at least an additional 60 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

SR 3.8.3.5

b. ~~By removing~~ Check for and remove accumulated water:

See ITS 3.8.1

1. From the engine-mounted fuel tank in accordance with the Surveillance Frequency Control Program and after each occasion when the diesel is operated for greater than 1 hour, and

SR 3.8.3.5

2. From ~~the~~ each fuel oil storage tank in accordance with the Surveillance Frequency Control Program.

See ITS 3.8.1

** The diesel generator start (10 sec.) from ambient conditions shall be performed in accordance with the Surveillance Frequency Control Program in these surveillance tests. All other diesel generator starts for the purposes of this surveillance testing may be preceded by an engine prelube period and may also include warmup procedures (e.g., gradual acceleration) as recommended by the manufacturer so that mechanical stress and wear on the diesel generator is minimized.

*** The indicated load band is meant as guidance to avoid routine overloading. Variations in loads in excess of the band due to changing bus loads shall not invalidate this test.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.3.3

c. Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of the Diesel Fuel Oil Testing Program.

d. DELETED

e. In accordance with the Surveillance Frequency Control Program by:

1. DELETED

2.

NOTE

Credit may be taken for unplanned events that satisfy this SR.

Verifying generator capability to reject the single largest post-accident load while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz.

3.

NOTE

This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

Verifying that upon an actual or simulated loss of offsite power signal by itself.

a) Deenergization of the emergency busses and load shedding from the emergency busses.

See ITS 3.8.1

Add proposed SR 3.8.3.4

M01

ELECTRICAL POWER SYSTEMS

SHUTDOWN

See ITS 3.8.2

LIMITING CONDITION FOR OPERATION

Add proposed LCO 3.8.3

A02

LCO 3.8.3 3.8.1.2 ~~As a minimum, the following A.C. electrical power sources shall be OPERABLE:~~

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator set with:

1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

SR 3.8.3.1

2. A fuel storage system containing a minimum of ~~19,000 gallons~~ a 7 day supply of fuel, and

LA01

See ITS 3.8.2

3. A fuel transfer pump.

Applicability **APPLICABILITY**: ~~MODES 5 and 6~~ When associated DG is required to be OPERABLE.

A02

A03

ACTIONS **ACTION**:

Add proposed ACTIONS Note

Add proposed ACTION A

Add proposed ACTIONS C and D

L01

L02

NOTE

Enter the ACTION of LCO 3.8.2.2, “A.C. Distribution - Shutdown,” with one required train de-energized as a result of inoperable offsite circuit.

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, movement of irradiated fuel, or crane operation with loads over the fuel storage pool. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the top of irradiated fuel assemblies seated within the reactor vessel, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

See ITS 3.8.2

SURVEILLANCE REQUIREMENTS

4.8.1.2.1 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2a.5.

See ITS 3.8.2

Add proposed ACTIONS B and E

M01

M01

L01

L02

Add proposed ACTION F

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

See ITS 3.8.1

Add proposed LCO 3.8.3

A02

LCO 3.8.3 3.8.1.1 ~~As a minimum, the following A.C. electrical power sources shall be OPERABLE:~~

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. Two separate engine-mounted fuel tanks containing a minimum volume of 238 gallons of fuel each,

SR 3.8.3.1

2. A separate fuel storage system containing a minimum volume of ~~42,500 gallons~~ of fuel, and

LA01

≥ a 7 day supply

3. A separate fuel transfer pump.

See ITS 3.8.1

A02

Applicability **APPLICABILITY**: ~~MODES 1, 2, 3, and 4~~.

When associated DG is required to be OPERABLE

A03

ACTIONS **ACTION**:

Add proposed ACTIONS Note

Add proposed ACTION A

Add proposed ACTIONS C and D

L01

L02

a. With one offsite circuit of 3.8.1.1.a inoperable, except as provided in ACTION f below:

1. Demonstrate the OPERABILITY of the remaining A. C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter.

2. Within 24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s), declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.

3. Restore the offsite circuit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

4. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.8.1

Add proposed ACTIONS B and E

M01

Add proposed ACTION F

M01

L01

L02

ELECTRICAL POWER SYSTEMS

See ITS 3.8.1

ACTION: (Continued)

d. With two of the required offsite A.C. circuits inoperable:

1. Within 12 hours from discovery of two offsite circuits inoperable concurrent with inoperability of redundant required feature(s), declare required feature(s) inoperable when its redundant required feature(s) is inoperable.

2. Restore one of the inoperable offsite sources to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours.

3. Following restoration of one offsite source, follow ACTION a with the time requirement of that ACTION based on the time of the initial loss of the remaining inoperable offsite A.C. circuit.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in the at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN. Following restoration of one diesel generator unit, follow ACTION Statement b. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator.

f. With one Unit 2 startup transformer (2A or 2B) inoperable and with a Unit 1 startup transformer (1A or 1B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 1 require the use of the startup transformer administratively available to both units, Unit 2 shall demonstrate the operability of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

g. LCO 3.0.4.b is not applicable to diesel generators.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments, indicated power availability; and

b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

SR 3.8.3.1

a. In accordance with the Surveillance Frequency Control Program by:

See ITS 3.8.1

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying fuel level in the engine-mounted fuel tank,

See ITS 3.8.1

SR 3.8.3.1

2. Verifying the fuel level in the fuel storage tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the engine-mounted tank,

4. Verifying the diesel starts from ambient condition and accelerates to approximately 900 rpm in less than or equal to 10 seconds**. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal**. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual/Local.

   b) Simulated loss-of-offsite power by itself.

   c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.

   d) An ESF actuation test signal by itself.

5. Verifying the generator is synchronized, loaded to greater than or equal to 3685 kW in accordance with the manufacturer's recommendations, and operates within a load band of 3450 to 3685 kW*** for at least an additional 60 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

Check for and remove

SR 3.8.3.5

b. ~~By removing~~ accumulated water:

See ITS 3.8.1

1. From the engine-mounted fuel tank in accordance with the Surveillance Frequency Control Program and after each occasion when the diesel is operated for greater than 1 hour, and

SR 3.8.3.5

each fuel oil

2. From ~~the~~ storage tank in accordance with the Surveillance Frequency Control Program.

See ITS 3.8.1

** The diesel generator start (10 sec.) from ambient conditions shall be performed in accordance with the Surveillance Frequency Control Program in these surveillance tests. All other diesel generator starts for purposes of this surveillance testing may be preceded by an engine prelube period and may also include warmup procedures (e.g., gradual acceleration) as recommended by the manufacturer so that mechanical stress and wear on the diesel generator is minimized.

*** The indicated load band is meant as guidance to avoid routine overloading. Variations in loads in excess of the band due to changing bus loads shall not invalidate this test.

Add proposed SR 3.8.3.2

M01

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.3

c. Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of the Diesel Fuel Oil Testing Program.

d. DELETED

e. In accordance with the Surveillance Frequency Control Program by:

1. DELETED

2.

NOTE
Credit may be taken for unplanned events that satisfy this SR.

Verifying generator capability to reject the single largest post-accident load while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz.

3.

NOTE
Credit may be taken for unplanned events that satisfy this SR.

Verifying the generator capability to reject a load of 3685 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection.

See ITS 3.8.1

Add proposed SR 3.8.3.4

M01

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO 3.8.3

3.8.1.2 ~~As a minimum, the following A.C. electrical power sources shall be OPERABLE:~~

Add proposed LCO 3.8.3

A02

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Two engine-mounted fuel tanks containing a minimum volume of 238 gallons of fuel,

SR 3.8.3.1

2. A fuel storage system containing a minimum volume of ~~42,500 gallons~~ of fuel, and

≥ a 7 day supply

See ITS 3.8.2

LA01

3. A fuel transfer pump.

See ITS 3.8.2

Applicability **APPLICABILITY**: ~~MODES 5 and 6~~.

When associated DG is required to be OPERABLE

A02

A03

ACTIONS **ACTION**:

Add proposed ACTIONS Note

Add proposed ACTION A

Add proposed ACTIONS C and D

L01

L02

NOTE

Enter the ACTION of LCO 3.8.3.2, “Onsite Power Distribution - Shutdown,” with one required train de-energized as a result of inoperable offsite circuit.

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to 3.58 square inch vent. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

See ITS 3.8.2

SURVEILLANCE REQUIREMENTS

4.8.1.2.1 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 (except for requirement 4.8.1.1.2a.5).

Add proposed ACTIONS B and E

M01

M01

L01

Add proposed ACTION F

L02

DISCUSSION OF CHANGES
ITS 3.8.3, DIESEL FUEL OIL, LUBE OIL, AND STARTING AIR

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS LCOs 3.8.1.1 and 3.8.1.2 state the requirements for the AC Sources during operating and shutdown conditions, respectively. These requirements are used to form the LCO and Applicability for the ITS diesel fuel oil Specification. ITS LCO 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," states that the stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required DG. The Applicability for this requirement is when the associated DG is required to be OPERABLE. This changes the CTS by combining the requirements for diesel fuel oil and diesel lube oil and starting air subsystem into one Specification.

This change is acceptable because the current requirements are translated into ITS form with no technical changes. The additional requirements for DG lube oil and starting air subsystem are discussed in DOC M01. Diesel fuel oil, lube oil and starting air subsystem are a support system for each DG. The CTS and ITS maintain this relationship between the DGs and the Diesel Fuel Oil System, DG Lube oil System and Starting Air Subsystem. This change is designated as administrative because it does not result in a technical change to the CTS.

A03 CTS 3.8.1.1 and 3.8.1.2 Actions, in part, provide requirements to be taken for each inoperable DG. ITS 3.8.3 ACTIONS include an explicit Note that states, "Separate Condition entry is allowed for each DG." This Note provides instructions for the proper application of the ACTIONS for ITS compliance. This changes the CTS by providing explicit direction for using the ACTIONS when diesel fuel oil and diesel lube oil and starting air subsystem parameters for a DG are not within limits.

This change is acceptable because the addition of the Note reflects the CTS allowance to take appropriate Actions for each DG. This change is designated as administrative since it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 The CTS does not provide any requirements for DG lube oil or starting air pressure. ITS LCO 3.8.3, in part, requires lube oil and starting air to be within limits for each required DG with associated Required Actions and Completion times when outside these limits. ITS SR 3.8.3.2 requires verification that the lube oil inventory is ≥ a 7-day supply for each DG. ITS SR 3.8.3.4 requires verification that required air start receiver bank pressure is ≥ 135 psig for each Unit 1 DG and

≥ 129 psig for each Unit 2 DG. ITS 3.8.3 ACTIONS B and E provide Required Actions if the required DG lube oil inventory is not within limits or the required starting air receiver bank pressure is not within limits respectively. This changes the CTS by adding limits for lube oil inventory and starting air, which Surveillance Requirements to verify the limits are being maintained and explicit ACTIONS for when the limits are not met.

The purpose of ITS 3.8.3 is to provide appropriate limits, ACTIONS, and Surveillances for diesel fuel oil, lube oil, and the air start subsystems because these subsystems support the operation of the standby AC power sources (DGs). The purpose of ITS SR 3.8.3.2 is to provide the appropriate limits for stored lube oil volume. ITS 3.8.3 ACTION B provides an explicit Required Action and Completion Time for restoring stored lube oil volume to within limits. The purpose of ITS SR 3.8.3.4 is to provide the appropriate limits for the DG starting air subsystem. ITS 3.8.3 ACTIONS E provides explicit Required Action and Completion Time for restoring starting air pressures to within limits. If a Required Action and associated Completion Time is not met or one or more DGs with lube oil inventory or required starting air receiver bank pressure not within limits for reasons other than addressed by Conditions B or E, ITS 3.8.3 ACTION F requires the associated DG to be declared inoperable immediately. This change is designated as more restrictive because explicit requirements are included in the Technical Specifications for lube oil inventory and starting air subsystems.

M02 **Unit 1 only:** CTS 3.8.1.1.b.2 and 3.8.1.2.b.2 require, for each DG set, the fuel oil storage system to contain a minimum of 19,000 gallons of fuel oil. ITS SR 3.8.3.1 requires a combined fuel inventory in the fuel oil storage tanks of ≥ a 7 day supply (See LA01 for discussion of relocating fuel oil volume value to the ITS Bases). The ITS Bases for SR 3.8.3.1 indicates that the fuel oil volume equivalent to a 7 day supply is 38,000 gallons. This changes the CTS by requiring more fuel oil volume be available to a DG to ensure at least 7 days of fuel oil is available to a DG.

The purpose of the ISTS minimum fuel oil storage requirement is to ensure that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at post accident load conditions. PSL Unit 1 is designed and licensed with a combined fuel oil storage inventory between two storage tanks to ensure one DG can operate for 7 days at post accident load conditions. These two fuel oil storage tanks are designed to supply each DG. Excess fuel oil in the Unit 2 fuel oil storage tanks can also be provided to support the Unit 1 DGs via a fuel transfer crosstie line between the unit’s storage tanks. A fuel oil level of 38,000 gallons is calculated in accordance with Regulatory Guide 1.137 and ANSI N195-1976 and takes into consideration a consumption of 36,107 gallons (7 days post-accident minus the usable day tank inventory), unusable storage tank volume, and assumed instrument tolerance.

This change is designated as more restrictive because the minimum fuel oil storage volume required to be maintained for a DG is increased from the value in CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.8.1.1.b.2 and CTS 3.8.1.2.b.2 require a separate fuel storage system containing a minimum volume of 19,000 gallons of fuel in each fuel storage tank for Unit 1 and 42,500 gallons in each fuel storage tank for Unit 2. Unit 1 ITS SR 3.8.3.1 requires verifying a combined fuel inventory in the fuel oil storage tanks is ≥ a 7-day supply (38,000 gals) when the diesel generator sets are required to be OPERABLE (See Discussion of Change M02 for change to fuel oil volume). Unit 2 ITS SR 3.8.3.1 requires, in part, verifying that each fuel oil storage tank contains ≥ a 7-day supply of fuel oil (42,500 gals) when the associated diesel generator is required to be OPERABLE. This changes the CTS by moving the specific values for the fuel oil inventory to the Bases and clarifying the Unit 1 fuel storage requirement to meet the 7 day criteria.

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. The purpose of CTS 3.8.1.1.b.2 and CTS 3.8.1.2.b.2 is to ensure that the DGs have sufficient fuel oil to perform their safety function when the DGs are required to be OPERABLE. ITS SR 3.8.3.1 will continue to ensure that sufficient fuel is contained in the fuel oil storage system for the DGs to perform their intended safety function when the DGs are required to be OPERABLE. In addition, this change is acceptable because these type of details will be adequately controlled in the Technical Specification Bases. Changes to the Bases are controlled by the Technical Specification Control Program in Chapter 5. The program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as less restrictive removal of detail change because details are being moved from the Technical Specifications to the ITS Bases

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* CTS 3.8.1.1 and CTS 3.8.1.2 do not provide explicit compensatory actions if the volume of fuel oil in a storage tank is less than the specified limit. Thus, if the minimum required volume is not met, the associated DG must be declared inoperable and the actions of CTS 3.8.1.1 or CTS 3.8.1.2 must be entered, as appropriate. ITS 3.8.3 ACTION A allows for not declaring the associated DG inoperable as long as the volume of stored fuel oil is greater than a 6 day supply. ITS 3.8.3 Required Action A.1 allows 48 hours to restore the fuel oil volume to within limits. If this Required Action and associated Completion Time is not met or if the fuel oil storage tank volume is < a 6-day supply, the associated DG(s) must be declared inoperable immediately, as required by ITS 3.8.3 ACTION F. For Unit 1, this would require declaring both DGs inoperable because the Unit 1 DGs rely on a combined fuel oil inventory volume in the fuel oil storage tanks (See Discussion of Change

M02). This changes the CTS by allowing the DGs to not be declared inoperable with the fuel oil less than a 7-day supply as long as the associated DG(s) has enough fuel oil for 6 days of operation at post accident load.

The purpose of ITS 3.8.3 ACTION A is to allow time to restore the stored diesel fuel oil volume to within the specified limit for a given DG. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to a degraded condition in order to minimize risk associated with continued operation while providing time to restore inventory. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features including the capacity and capability of remaining systems or features, reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. The addition of ITS 3.8.3 ACTION A will allow each DG to not be declared inoperable with the associated stored diesel fuel oil volume not within the specified Surveillance limit as long as the DG has enough fuel oil for 6 days of operation at post accident load. In this Condition, the 7 day fuel oil supply for the DG is not available. However, the Condition is restricted to fuel oil volume that maintains at least a 6 day supply. These circumstances may be caused by events such as full load operation required after an inadvertent start while at minimum required fuel oil level, or feed and bleed operations, which may be necessary due to increasing particulate levels or any number of other oil quality degradations. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the associated DG(s) inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period. This change is designated as less restrictive since less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 4.8.1.1.2.c requires, in part, to verify fuel oil properties of new and stored fuel oil are tested in accordance with and maintained within limits specified in the Diesel Fuel Oil Testing Program. CTS 6.8.4.n provides requirements for new fuel oil properties and fuel oil total particulates. CTS 3.8.1.1 and CTS 3.8.1.2 do not provide explicit compensatory actions if the new fuel oil properties or fuel oil total particulates are not within the limits specified in the Diesel Fuel Oil Testing Program. Thus, if the fuel oil properties are not met, the associated DG must be declared inoperable and the actions of CTS 3.8.1.1 or CTS 3.8.1.2 must be entered, as appropriate. ITS SR 3.8.3.3 also requires verification that new and stored fuel oil properties are tested and maintained within limits, as specified in the Diesel Fuel Oil Testing Program. This includes a stored fuel oil total particulate limit and new fuel oil properties. ITS 3.8.3 ACTION C specifies the compensatory actions for one or more DGs with stored fuel oil total particulates not within limits. ITS 3.8.3 ACTION D specifies the compensatory actions for one or more DGs with new fuel oil properties not within limits. If a Required Action and associated Completion Time is not met or one or more DGs with new and stored fuel oil properties not within limits for reasons other than addressed by Conditions C or D, ITS 3.8.3 ACTION F requires the associated DG to be declared inoperable immediately. This

changes the CTS by allowing the DGs to not be declared inoperable with fuel oil total particulates or new fuel oil properties not within the specified Surveillance limits unless the applicable ITS 3.8.3 ACTIONS cannot be completed within the required Completion Time.

The purpose of CTS 4.8.1.1.2.c (ITS SR 3.8.3.3) is to provide the appropriate limits for stored and new fuel oil properties. ITS 3.8.3 ACTIONS C and D provide explicit Required Actions and Completion Times for restoring both total particulates and stored fuel oil properties (affected by the addition of new fuel oil whose properties are not within limits) to within limits. ITS 3.8.3 ACTION C is entered when the acceptance criterion of total particulate concentration specified in ITS 5.5.10 is not met. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures, contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not necessarily result in a failure of the fuel oil to burn properly in the diesel engine, particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, re-sampling and re-analysis of the DG fuel oil. ITS 3.8.3 ACTION D is entered because of failure to meet the requirements specified in ITS 5.5.10.b. With the new fuel oil properties defined in the Bases for ITS SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil did not cause the stored fuel oil to be outside of the required limits, or to restore the stored fuel oil properties to within limits. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the stored fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function. ITS 3.8.3 ACTION F covers the condition when a Required Action and associated Completion Time of any ITS 3.8.3 ACTIONS could not be met, or if one or more DGs with diesel fuel oil, lube oil, or starting air subsystem is not within limits for reasons other than Condition A, B, C, D, or E. The allowances are considered acceptable since a relatively short time is allowed to exist with DG parameters not within limits

This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a design basis accident occurring during the repair period. The fuel oil properties must be restored within limits within 30 days and the fuel oil total particulates must be stored in 7 days. In this condition, the DG fuel oil is degraded but remains capable of supporting the DG OPERABILITY. The ITS

requirements are consistent with the ISTS wording for these requirements. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

3.8.1.1
3.8.1.2

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

Applicability

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

----------NOTE----------

DOC A03

Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 A. One or more DGs with ~~fuel level~~ less than a [7] day supply and greater than a [6] day supply in storage tank(s).	A.1 Restore fuel oil level to within limits.	48 hours (1) (2)
DOC M01 B. One or more DGs with lube oil inventory less than a [7] day supply and greater than a [6] day supply.	B.1 Restore lube oil inventory to within limits.	48 hours (1)
DOC L02 C. One or more DGs with stored fuel oil total particulates not within limits.	C.1 Restore fuel oil total particulates to within limits.	7 days
DOC L02 D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01 E. One or more DGs with required starting air receiver bank 135 pressure < [225] psig and 85 ≥ [125] psig.	E.1 135 Restore starting air receiver pressure to ≥ [225] psig.	48 hours (2, 1)
DOC M01 DOC L01 DOC L02 F. Required Action and associated Completion Time not met. OR One or more DGs with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.1.1.2.a.2 3.8.1.1.b.2 3.8.1.2.b.2 DOC M02 SR 3.8.3.1	Verify ~~each~~ fuel oil storage tank ~~contains~~ (s is) ≥ a [7] day supply ~~of fuel~~ (combined fuel inventory in).	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2, 1)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
DOC M01	SR 3.8.3.2	Verify lubricating oil inventory is ≥ a ~~[~~7~~]~~ day supply.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1
4.8.1.1.2.c	SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program	
DOC M01	SR 3.8.3.4	Verify each DG air start receiver pressure is ≥ ~~[225]~~ 135 psig. (required; bank)	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	2 1
4.8.1.1.2.b.2	SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	~~[[31] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

3.8.1.1
3.8.1.2

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

Applicability

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

----------NOTE----------

DOC A03

Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 A. One or more DGs with fuel level less than a ~~[~~7~~]~~ day supply and greater than a ~~[~~6~~]~~ day supply in storage tank.	A.1 Restore fuel oil level to within limits.	48 hours (1)
DOC M01 B. One or more DGs with lube oil inventory less than a ~~[~~7~~]~~ day supply and greater than a ~~[~~6~~]~~ day supply.	B.1 Restore lube oil inventory to within limits.	48 hours (1)
DOC L02 C. One or more DGs with stored fuel oil total particulates not within limits.	C.1 Restore fuel oil total particulates to within limits.	7 days
DOC L02 D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M01 E. One or more DGs with required starting air receiver bank pressure < [225] psig and ≥ [125] psig.	E.1 Restore starting air receiver pressure to ≥ [225] psig.	48 hours
DOC M01 DOC L01 DOC L02 F. Required Action and associated Completion Time not met. OR One or more DGs with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.1.1.2.a.2 3.8.1.1.b.2 3.8.1.2.b.2 SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ a [7] day supply of fuel.	[31 days OR In accordance with the Surveillance Frequency Control Program]

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
DOC M01	SR 3.8.3.2	Verify lubricating oil inventory is ≥ a ~~[~~7~~]~~ day supply.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1
4.8.1.1.2.c	SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program	
DOC M01	SR 3.8.3.4	Verify each DG air start receiver pressure is ≥ ~~[225]~~ psig. (required; bank; 129)	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	2 1
4.8.1.1.2.b.2	SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	~~[[31] days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.3, DIESEL FUEL OIL, LUBE OIL, AND STARTING AIR

1. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

BASES

BACKGROUND

~~Each diesel generator (DG) is~~ provided with a ~~storage tank~~ having a fuel oil capacity sufficient to operate ~~that diesel~~ for a period of ~~[~~7~~]~~ days, while the DG is supplying ~~maximum~~ post loss of coolant accident load demand as discussed in the FSAR, Section ~~[~~9.5.4.2~~]~~ (Ref. 1) ~~[~~and Regulatory Guide 1.137 (Ref. 2)~~]~~. The ~~maximum~~ load demand is calculated using the assumption that ~~at least two~~ DG~~s~~ ~~are~~ available. This onsite fuel oil capacity is sufficient to operate the DG~~s~~ for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from storage tank to day tank by ~~either of two~~ transfer pump~~s~~ associated with each storage tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank to result in the loss of more than one DG. All ~~outside tanks, pumps, and piping are~~ located underground.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195-1976 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG ~~under all~~ loading conditions. The system is required to circulate the lube oil to the ~~diesel engine~~ working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting ~~a minimum of [7] days of~~ operation. ~~[~~The onsite storage in addition to ~~the~~ engine oil sump is sufficient to ensure ~~[~~7~~]~~ days of continuous operation.~~]~~ This supply is sufficient ~~supply~~ to allow the operator to replenish lube oil from outside sources.

Each DG has an air start system with adequate capacity for five ~~successive~~ start ~~attempts~~ on the DG without recharging the air start receiver(s).

The diesel generators (DGs) are
fuel supply system
one DG
profile
U
Each DG consists of two tandem diesel engines and a generator.
one
is
Insert 1
post accident
1
2
1
The fuel oil storage system consists of two subsystems with each subsystem consisting of a fuel oil storage tank, transfer pump, two day tanks, interconnecting piping and valves.
a
1
Cross connection piping is provided with locked closed valves for transferring fuel oil between the redundant subsystems.
The fuel oil storage tanks are located on an outdoor platform.
fuel oil piping, with the exception of the suction and discharge cross-tie lines, is routed through enclosed buildings or
engines under post accident
of the diesel engines
each
1
2
3
two redundant subsystems each with
cold
s
per air start receiver bank
1
One DG air start subsystem supplies air simultaneously to air start motors of both tandem diesel engines during a DG start.

1

INSERT 1

Sufficient onsite fuel oil storage is provided to support 7 days of DG operation for one DG at post accident load conditions. Onsite fuel oil storage inventory includes the volume of both Unit 1 fuel oil storage tanks and excess fuel in the Unit 2 fuel oil storage tanks. Fuel oil transfer capability is provided between the storage tanks to ensure at least one DG has a 7-day supply of fuel.

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 4), and in the FSAR, Chapter ~~[~~15~~]~~ (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for LCO Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, and the air start subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Stored diesel fuel oil is required to have sufficient supply for ~~[~~7~~]~~ days ~~of full~~ load operation. ~~It is~~ also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate at full load for ~~[~~7~~]~~ days. This requirement, in conjunction with an ability to obtain replacement supplies within ~~[~~7~~]~~ days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown."

The starting air system is required to have a minimum capacity for five ~~successive~~ DG start~~attempts~~ without recharging the air start receivers.

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil, lube oil, and starting air subsystems support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

BASES

ACTIONS (continued)

A.1

In this Condition, ~~the~~ [~~7~~] day ~~fuel oil supply~~ for a DG is not available. However, the Condition is restricted to fuel oil level reductions, that maintain at least a [~~6~~] day ~~supply~~. The fuel oil ~~level~~ equivalent to a [~~6~~] day supply is [~~28,285~~] gallons. These circumstances may be caused by events such as full load operation required after an inadvertent start while at minimum required level; or feed and bleed operations, which may be necessitated by increasing particulate levels or any number of other oil quality degradations. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining ~~capacity~~ (≥ [~~6~~] days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

a

combined fuel oil inventory in the fuel oil storage tanks

to support either DG

volume

5

2

32,572

combined onsite fuel oil storage tank inventory

2

1

If the combined fuel oil inventory in the fuel oil storage tanks is < 6 days or a combined inventory of 7 days is not restored within 48 hours, both DGs must be declared inoperable per Required Action F.1.

5

B.1

In this Condition, the [~~7~~] day lube oil inventory i.e., sufficient lubricating oil to support [~~7~~] days of continuous DG operation at ~~full~~ load conditions is not available. However, the Condition is restricted to lube oil volume reductions that maintain at least a [~~6~~] day supply. The lube oil inventory equivalent to a [~~6~~] day supply is [~~425~~] gallons. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> [~~6~~] days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

onsite (500 gallons)

1

post accident

428

2

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.5. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change

BASES

ACTIONS (continued)

significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.4 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

required

135

With starting air receiver pressure < ~~[225]~~ psig, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver pressure is > ~~[125]~~ psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

85

1

2

F.1

With a Required Action and associated Completion Time not met, or one or more DGs with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

post accident load conditions. Implicit in this SR is the requirement to verify the capability to transfer the fuel oil from the other onsite storage tanks to the DG day tank

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

volume

one

combined onsite

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support ~~each~~ DG's operation for [7] days at ~~full load~~. The fuel oil ~~level~~ equivalent to a [7] day supply is [~~33,~~38,000] gallons when calculated in accordance with References 2 and 3. The required fuel storage volume is determined using the most limiting energy content of the stored fuel. Using the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel generator output, and the corresponding fuel consumption rate, the onsite fuel storage volume required for [7] days of operation can be determined. SR 3.8.3.3 requires new fuel to be tested to verify that the absolute specific gravity or API gravity is within the range assumed in the diesel fuel oil consumption calculations. The [7] day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

38,000

The calculated value takes into consideration a consumption of 36,107 gallons (7 days post-accident minus the usable day tank inventory), unusable storage tank volume, and assumed instrument tolerance.

2 1 5 2 2

[~~The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

] 2 4 2

SR 3.8.3.2

onsite

post accident

This Surveillance ensures that sufficient lube oil inventory is available to support at least [7] days of ~~full~~ load operation for each DG. The lube oil inventory equivalent to a [7] day supply is [500] gallons and is based on the DG manufacturer consumption values for the run time of the DG. Implicit in this SR is the requirement to verify the capability to transfer the lube oil from its storage location to the DG, when the DG lube oil ~~sump does~~ not hold adequate inventory for [7] days of full load operation without the level reaching the manufacturer recommended minimum level.

1 2 1 2

sumps do

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the unit staff.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057-~~[ ]~~ (Ref. 6),

b. Verify in accordance with the tests specified in ASTM D975-~~[ ]~~ (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89, or an API gravity at 60°F of ≥ 27° and ≤ 39° when tested in accordance with ASTM D1298-~~[ ]~~ (Ref. 6), a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point ≥ 125°F, and

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-~~[ ]~~ or a water and sediment content within limits when tested in accordance with ~~[~~ASTM D2709-~~[ ]]~~ (Ref. 6).

BASES

SURVEILLANCE REQUIREMENTS (continued)

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-~~[ ]~~ (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975-~~[ ]~~ (Ref. 6), except that the analysis for sulfur may be performed in accordance with ASTM D~~1552-[ ]~~, ASTM D2622-~~[ ]~~, or ASTM D~~4294-[ ] (Ref. 6)~~. The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

06
06
5453
3120
82
2
1

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D~~5452-[ ] (Ref. 6)~~. This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. ~~[For those designs in which the~~ total stored fuel oil volume ~~is~~ contained in two or more interconnected tanks~~, each tank must be~~ considered and tested separately.~~]~~

6217 or D2276
Each tank of the
when
is
1
2

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of ~~[~~five~~]~~ ~~engine~~ start ~~cycles~~ without recharging. ~~[A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds or cranking) or engine cranking speed.]~~ The pressure specified in this SR is intended to reflect the lowest value at which the ~~[~~five~~]~~ starts can be accomplished.

cold
of both tandem diesel engines
cold
s
1
2
1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. ~~[The Surveillance Frequency of 31 days is established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventative maintenance.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed during performance of the Surveillance.

REFERENCES

1. FSAR, Section ~~[~~9.5.4.2~~]~~.
2. Regulatory Guide 1.137.
3. ANSI N195, 1976.
4. FSAR, Chapter ~~[~~6~~]~~.
5. FSAR, Chapter ~~[~~15~~]~~.
6. ASTM Standards: D4057-~~[ ]~~; D975-~~[ ]~~; D1298-~~[ ]~~; D4176-~~[ ]~~; ~~[~~D2709-~~[ ];]~~ ~~D1552-[ ]~~; D2622-~~[ ]~~; ~~D4294-[]~~; D545~~2-[ ]~~; D3120; D2276-83.
7. ASTM Standards, D975-~~[ ]~~, Table 1.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

BASES

BACKGROUND

Each diesel generator (DG) is provided with a storage tank having a fuel oil capacity sufficient to operate that diesel for a period of ~~[~~7~~]~~ days, while the DG is supplying ~~maximum~~ post ~~loss of coolant~~ accident load demand as discussed in the FSAR, Section ~~[~~9.5.4.2~~]~~ (Ref. 1) ~~[~~and Regulatory Guide 1.137 (Ref. 2)~~]~~. The ~~maximum~~ load demand is calculated using the assumption that ~~at least two~~ DG~~s~~ ~~are~~ available. This onsite fuel oil capacity is sufficient to operate the DG~~s~~ for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from storage tank to day tank by ~~either of two~~ transfer pump~~s~~ associated with each storage tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank to result in the loss of more than one DG. All ~~outside tanks, pumps, and piping are~~ located underground.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195-1976 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG ~~under all~~ loading conditions. The system is required to circulate the lube oil to the ~~diesel engine~~ working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting ~~a minimum of [7] days of~~ operation. ~~[~~The onsite storage in addition to ~~the~~ engine oil sump is sufficient to ensure ~~[~~7~~]~~ days of continuous operation.~~]~~ This supply is sufficient ~~supply~~ to allow the operator to replenish lube oil from outside sources.

Each DG has an air start system with adequate capacity for five ~~successive~~ start ~~attempts~~ on the DG without recharging the air start receiver(s).

and two day tanks
1
profile
U
profile
2
1
A DG consists of two tandem diesel engines and a generator.
one
is
1
post accident
a
Cross connection piping is provided with locked closed valves for transferring fuel oil between fuel storage systems.
1
fuel oil piping is routed through enclosed buildings or
engines under post accident
of the diesel engines
1
2
each
3
cold
s
per air start receiver bank
1
The DG air start subsystem supplies air simultaneously to air start motors of both tandem diesel engines during a DG start.

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 4), and in the FSAR, Chapter [15] (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for LCO Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, and the air start subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Stored diesel fuel oil is required to have sufficient supply for [7] days of full load operation. It is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate at full load for [7] days. This requirement, in conjunction with an ability to obtain replacement supplies within [7] days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown."

The starting air system is required to have a minimum capacity for five successive DG start attempts without recharging the air start receivers.

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil, lube oil, and starting air subsystems support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

BASES

ACTIONS (continued)

A.1

In this Condition, the [7] day fuel oil supply for a DG (42,500) is not available. However, the Condition is restricted to fuel oil level reductions, that maintain at least a [6] day supply. The fuel oil level equivalent to a [6] day supply is 36,429 [~~28,285~~] gallons. These circumstances may be caused by events such as full load operation required after an inadvertent start while at minimum required level; or feed and bleed operations, which may be necessitated by increasing particulate levels or any number of other oil quality degradations. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (≥ [6] days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period. (1) (2) (2)

B.1

In this Condition, the [7] day lube oil inventory onsite (500 gallons) i.e., sufficient lubricating oil to support [7] days of continuous DG operation at ~~full~~ post accident load conditions is not available. However, the Condition is restricted to lube oil volume reductions that maintain at least a [6] day supply. The lube oil inventory equivalent to a [6] day supply is 428 [~~425~~] gallons. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> [6] days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period. (1) (2)

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.5. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change

BASES

ACTIONS (continued)

significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.4 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

required

129

With starting air receiver pressure < ~~[225]~~ psig, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver pressure is > ~~[125]~~ psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

85

2

F.1

With a Required Action and associated Completion Time not met, or one or more DGs with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for [7] days at full load. The fuel oil level equivalent to a [7] day supply is [33,000] gallons when calculated in accordance with References 2 and 3. The required fuel storage volume is determined using the most limiting energy content of the stored fuel. Using the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel generator output, and the corresponding fuel consumption rate, the onsite fuel storage volume required for [7] days of operation can be determined. SR 3.8.3.3 requires new fuel to be tested to verify that the absolute specific gravity or API gravity is within the range assumed in the diesel fuel oil consumption calculations. The [7] day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

post-accident load conditions

42,500

The calculated value takes into consideration a consumption of 40,671 gallons (7 days post-accident minus the usable day tank inventory), unusable storage tank volume, and assumed instrument tolerance.

2 1 5 2 2

[The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE

Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

]

2 4 2

SR 3.8.3.2

This Surveillance ensures that sufficient lube oil inventory is available to support at least [7] days of full load operation for each DG. The lube oil inventory equivalent to a [7] day supply is [500] gallons and is based on the DG manufacturer consumption values for the run time of the DG. Implicit in this SR is the requirement to verify the capability to transfer the lube oil from its storage location to the DG, when the DG lube oil sump does not hold adequate inventory for [7] days of full load operation without the level reaching the manufacturer recommended minimum level.

onsite

post-accident

sumps do

1 2 1 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the unit staff.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057-~~[ ]~~ (Ref. 6),

b. Verify in accordance with the tests specified in ASTM D975-~~[ ]~~ (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89, or an API gravity at 60°F of ≥ 27° and ≤ 39° when tested in accordance with ASTM D1298-~~[ ]~~ (Ref. 6), a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point ≥ 125°F, and

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-~~[ ]~~ or a water and sediment content within limits when tested in accordance with ~~[~~ASTM D2709-~~[ ]]~~ (Ref. 6).

BASES

SURVEILLANCE REQUIREMENTS (continued)

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

Within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-~~[ ]~~ (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975-~~[ ]~~ (Ref. 6), except that the analysis for sulfur may be performed in accordance with ASTM D~~1552-[ ]~~, ASTM D2622-~~[ ]~~, or ASTM D~~4294-[ ] (Ref. 6)~~. The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

06
06
5453
3120
82
2
1

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D~~5452-[ ] (Ref. 6)~~. This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. ~~[For those designs in which the~~ total stored fuel oil volume ~~is~~ contained in two or more interconnected tanks~~, each tank must be~~ considered and tested separately.~~]~~

6217 or D2276
Each tank of the
when
is
1
2

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of ~~[~~five~~]~~ ~~engine~~ start ~~cycles~~ without recharging. ~~[A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds or cranking) or engine cranking speed.]~~ The pressure specified in this SR is intended to reflect the lowest value at which the ~~[~~five~~]~~ starts can be accomplished.

cold
of both tandem diesel engines
cold
s
1
2
1

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. ~~[The Surveillance Frequency of 31 days is established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventative maintenance.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 4 2

The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed during performance of the Surveillance.

REFERENCES

1. FSAR, Section ~~[~~9.5.4.2~~]~~. U 1 2
2. Regulatory Guide 1.137.
3. ANSI N195, 1976.
4. FSAR, Chapter ~~[~~6~~]~~. U
5. FSAR, Chapter ~~[~~15~~]~~. 1 2
6. ASTM Standards: D4057-~~[ ]~~; D975-~~[ ]~~; D1298-~~[ ]~~; D4176-~~[ ]~~; ~~[~~D2709-~~[ ];]~~ ~~D1552-[ ]~~; D2622-~~[ ]~~; ~~D4294-[]~~; D545~~2-[ ]~~ ; D3120; D2276-83.
7. ASTM Standards, D975-~~[ ]~~, Table 1.

81 06 99 82 82 06 6217 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.3 BASES, DIESEL FUEL OIL, LUBE OIL, AND STARTING AIR

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Editorial grammar changes to the Bases have been made to enhance clarity.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Changes are made to clarify the 7 day fuel oil storage volume requirement.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.8.3, DIESEL FUEL OIL, LUBE OIL AND STARTING AIR

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 4

ITS 3.8.4, DC Sources - Operating

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ELECTRICAL POWER SYSTEMS

SOURCES

D.C. ~~DISTRIBUTION~~ - OPERATING

LIMITING CONDITION FOR OPERATION

The Train A and Train B DC electrical power subsystems shall be OPERABLE.

LCO 3.8.4 3.8.2.3 ~~As a minimum the following D.C. electrical sources shall be OPERABLE:~~

a. 125-volt D.C. bus No. 1A, ~~125-volt Battery bank No. 1A and a full capacity charger.~~

b. 125-volt D.C. bus No. 1B, ~~125-volt Battery bank No. 1B and a full capacity charger.~~

See ITS 3.8.9

LA01

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTION:

See ITS 3.8.9

Action B — a. With one of the required battery banks or busses inoperable, restore the inoperable battery bank or bus to OPERABLE status within 2 hours or be in at

Action C — least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Required Action C.2 NOTE — LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Action A — b. With one of the required full capacity chargers inoperable, ~~demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.3.2.a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.~~

Add proposed Required Actions A.1, A.2 and A.3 and Completion Times

L01

M01

See ITS 3.8.9

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying indicated power availability.

4.8.2.3.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

See ITS 3.8.6

SR 3.8.4.1 2. The total battery terminal voltage is greater than or equal to ~~129 volts on float charge~~.

132 V

the minimum established float voltage

LA02

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits, — See ITS 3.8.6

2. There is no visible corrosion at either terminals or connectors, and — See ITS 5.5.14

3. The average electrolyte temperature of 10% (60 cells total) of connected cells is above 50°F. — See ITS 3.8.6

c. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3. Battery cell inter-connection resistance values are maintained at the values below: — See ITS 3.8.6

Battery Inter-Connection Measurement Limits		
Battery Inter-Connection Type	Maximum Individual Inter-Connection Resistance	Maximum Average Inter-Connection Resistance [Battery Bank*]
Inter-Cell	≤ 150 x 10-6 ohms	≤ 50 x 10-6 ohms
Inter-Tier	≤ 200 x 10-6 ohms	
Inter-Rack	≤ 200 x 10-6 ohms	
Output Terminal	≤ 150 x 10-6 ohms	

* The battery bank average interconnection resistance limit is the average of all inter-cell, inter-tier, inter-rack and output terminal connection resistance measurements for all series connections in the battery string

and,

SR 3.8.4.2

4. The battery charger will supply at least 300 amperes at 140 volts for at least 6 hours.

Add proposed 2nd option for SR 3.8.4.2 — L02

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

Add proposed SR 3.8.4.3 NOTE 2

L03

SR 3.8.4.3

d. In accordance with the Surveillance Frequency Control Program~~, during shutdown~~, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. ~~This~~ performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.3.2.d.

See ITS 3.8.6

SR 3.8.4.3 NOTE 1

Modified

M02

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

See ITS 3.8.6

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENT

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(c)	> 2.07 volts
Specific Gravity(a)	≥ 1.195(b)	≥ 1.190 Average of all connected cells > 1.200	Not more than .020 below the average of all connected cells Average of all connected cells ≥ 1.190(b)

See ITS 3.8.6

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amps when on charge.

(c) Corrected for average electrolyte temperature.

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within the limits within 7 days.

(3) With any Category B parameter not within its allowable value, declare the battery inoperable.

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

The Train A and Train B DC electrical power subsystems shall be OPERABLE.

LCO 3.8.4 3.8.2.1 ~~As a minimum the following D.C. electrical sources shall be OPERABLE:~~

a. ~~125-volt Battery bank No. 2A and a full capacity charger.~~

b. ~~125-volt Battery bank No. 2B and a full capacity charger.~~

LA01

Applicability **APPLICABILITY**: MODES 1, 2, 3, and 4.

ACTION:

Action B / Action C / Required Action C.2 NOTE

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Action A

b. With one of the required full capacity chargers inoperable, ~~demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.1a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.~~

Add proposed Required Actions A.1, A.2 and A.3 and Completion Times

L01

M01

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and — See ITS 3.8.6

SR 3.8.4.1

2. The total battery terminal voltage is greater than or equal to ~~129 volts on float charge~~.

132 V

the minimum established float voltage

LA02

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits, — See ITS 3.8.6

2. There is no visible corrosion at either terminals or connectors, and — See ITS 5.5.14

3. The average electrolyte temperature of 10% (60 cells total) of connected cells is above 50°F. — See ITS 3.8.6

c. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material, — See ITS 3.8.6

3. Battery cell inter-connection resistance values are maintained at the values below:

Battery Inter-Connection Measurement Limits		
Battery Inter-Connection Type	Maximum Individual Inter-Connection Resistance	Maximum Average Inter-Connection Resistance [Battery Bank*]
Inter-Cell	≤ 150 x 10-6 ohms	≤ 50 x 10-6 ohms
Inter-Tier	≤ 200 x 10-6 ohms	
Inter-Rack	≤ 200 x 10-6 ohms	
Output Terminal	≤ 150 x 10-6 ohms	

* The battery bank average interconnection resistance limit is the average of all inter-cell, inter-tier, inter-rack and output terminal connection resistance measurements for all series connections in the battery string

and,

SR 3.8.4.2

4. The battery charger will supply at least 300 amperes at 140 volts for at least 6 hours.

Add proposed 2nd option for SR 3.8.4.2 — L02

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

Add proposed SR 3.8.4.3 NOTE 2

L03

SR 3.8.4.3

d. In accordance with the Surveillance Frequency Control Program~~, during shutdown~~, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. ~~This~~ performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1d.

See ITS 3.8.6

M02

SR 3.8.4.3 NOTE 1

Modified

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

See ITS 3.8.6

TABLE 4 8-2

BATTERY SURVEILLANCE REQUIREMENT

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(c)	> 2.07 volts
			Not more than .020 below the average of all connected cells
		≥ 1.190	
Specific Gravity(a)	≥ 1.195(b)	Average of all connected cells > 1.200	Average of all connected cells ≥ 1.190(b)

See ITS 3.8.6

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amps when on charge.

(c) Corrected for average electrolyte temperature.

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within the limits within 7 days.

(3) With any Category B parameter not within its allowable value, declare the battery inoperable.

DISCUSSION OF CHANGES
ITS 3.8.4, DC SOURCES - OPERATING

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS LCO 3.8.2.3 and Unit 2 CTS LCO 3.8.2.1 Action b state, “With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.3.2.a.1 within one hour, and at least once per 8 hours thereafter.” ITS LCO 3.8.4 Condition A requires, in part, with the required battery charger on one subsystem inoperable, that the required inoperable charger must be restored to OPERABLE status within 72 hours. Refer to DOC L01 for discussion related to less restrictive actions. This changes the CTS by adding a requirement to restore the required charger to Operable status.

The purpose of ITS 3.8.4 Required Actions is to provide the appropriate actions to be taken when a charger becomes inoperable. This change is acceptable because the additional requirements are appropriate to ensure the battery is returned to a condition that it can provide the required safety functions. The ITS requirements are consistent with the ISTS wording for this requirement. This change is designated as more restrictive because it adds an additional required action to restore a required battery charger within 72 hours.

M02 Unit 1 CTS 4.8.2.3.2.e and Unit 2 4.8.2.1.e allow the performance discharge test to be performed in lieu of the battery service test. ITS SR 3.8.4.3 Note 1 provides a similar allowance but requires the performance discharge test to be a modified performance discharge test to be used in lieu of the battery services test. This changes the CTS by requiring a more restrictive type of performance discharge test to be performed in lieu of the battery service test.

The purpose of allowing a performance discharge test in lieu of a battery service test is to eliminate the need to perform two separate tests, thus minimizing the time the battery is unavailable to perform its related electrical support function.

This change is necessary because the normal performance discharge test is inadequate to satisfy the battery service test requirements. A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle) and is described in IEEE 450, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications.” The modified performance discharge test combines the greater of both the peak duty cycle load values of the service test with the load values and duration of the

performance capacity test. This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test. It may consist of just two rates; for instance the one minute rate for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test must remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test. The ITS requirements are consistent with the ISTS wording for this requirement. This change is designated as more restrictive because it requires a more restrictive allowance than CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS LCO 3.8.2.3 and Unit 2 CTS LCO 3.8.2.1 state, in part, that the DC electrical sources shall be OPERABLE and lists the details of what constitutes a DC electrical power source. ITS LCO 3.8.4 requires the Train A and Train B DC electrical power subsystems to be OPERABLE. This changes the CTS by moving the details of the components of the DC Sources (battery and charger) from the CTS to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirements for the DC electrical power subsystems. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Control Program in Chapter 5. The program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information related to system design is being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS surveillance requirement 4.8.2.3.2.a.2 and Unit 2 surveillance requirement 4.8.2.1.a.2 state that each 125 volt battery bank and charger shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program with total battery terminal voltage greater than or equal to 129 volts on float charge. ITS SR 3.8.4.1 requires verification that battery terminal voltage is greater than or equal to “the minimum established float voltage.” This changes the CTS by moving the battery terminal voltage requirements from the specifications to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The Technical Specifications still retain the surveillance requirement to verify battery terminal voltage is ≥ the minimum established float voltage. The minimum establish float voltage for the PSL batteries is 2.20 Vpc as specified by the battery manufacturer and equates to 132 V for a 60 cell battery. As such, the 129 volts is changed to 132 V in the ITS Bases to reflect the minimum float voltage specified by the battery manufacturer. This change is acceptable because the removed information will be adequately controlled in the Technical Specification Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Section 5 of the Technical Specifications. This program provides for the evaluation of Bases changes in accordance with 10 CFR 50.59 to ensure the Bases are properly controlled. The ITS requirement is consistent with the ISTS wording for this requirement. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 (Category 4 – Relaxation of Required Action) Unit 1 CTS 3.8.2.3 and Unit 2 CTS 3.8.2.1 Action b in part states with one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.3.2.a.1 (Unit 1) and Surveillance Requirement 4.8.2.1.a.1 (Unit 2) within 1 hour. The requirement goes on to state that the action requires the applicable Surveillance Requirement to be performed once per 8 hours thereafter and if any Category A limit in Table 3.8-1 is not met, declare the battery inoperable. ITS LCO 3.8.4 Condition A states, in part, with one required battery charger on one train inoperable, restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours (Required Action A.1). Required Action A.2 states a verification of float current of ≤ 2 amps is required once per 12 hours. This changes the CTS by allowing the battery terminal voltage to be restored to the minimum established float voltage with float current to be ≤ 2 amps instead of declaring the battery inoperable based on individual cell parameters.

The purpose of ITS Required Actions is to allow appropriate remedial actions with appropriate time limitations prior to restoring an inoperable battery charger to OPERABLE status. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a design basis accident occurring during the repair period. This change establishes the appropriate actions to be taken with the required battery charger inoperable. The current requirement specifies that all Category A parameters listed in Table 3.8-1

be within limits. The battery charger affect on the associated battery is directly related to the terminal voltage and the charging current in a float condition and not necessarily individual battery cell parameters. With a charger with the capability of restoring the minimum float voltage and less than 2 amps of charging current, the battery can continue to perform its safety related support function and should be considered OPERABLE. Twelve hours is acceptable for the verification of the charging current because it provides a limited time and the required charger must be returned to OPERABLE within 72 hours (Refer to DOC M02 for discussion of restoring the required battery charger). The ITS requirement is consistent with the ISTS wording for this requirement. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS surveillance requirement 4.8.2.3.2.c.4 and Unit 2 surveillance requirement 4.8.2.1.c.4 state that the battery charger will supply at least 300 amps at 140 volts for at least 6 hours. ITS SR 3.8.4.2 provides an alternative test method that allows a verification of each battery charger can recharge the battery to the fully charged state within 12 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state. This changes the CTS by allowing an alternate test that is not currently allowed. | R1

The purpose of Unit 1 CTS 4.8.2.3.2.c.4 and Unit 2 surveillance requirement 4.8.2.1.a.2 is to verify the required 125 V DC battery chargers can recharge their respective batteries following a loss of offsite power event. This change is acceptable because the relaxed Surveillance Requirement acceptance criteria are adequate to verify that the equipment used to meet the LCO can perform its required functions. The alternate test provides an acceptable method for determining charger capability by actually recharging a discharged battery within 12 hours while supplying required loads. The ITS requirement is consistent with the ISTS wording for this requirement. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L03 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.8.2.3.2.d and Unit 2 4.8.2.1.d state that battery capacity is verified by subjecting the battery to a service test. ITS SR 3.8.4.3 requires a service test to be performed to verify the battery capacity with a modifying Note 2. Note 2 that modifies the SR states “This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.” This changes the CTS by allowing portions of the SR to be performed in MODES 1, 2, 3, or 4 and allowing credit for unplanned events.

The purpose of ITS SR 3.8.4.3 Note 2 is to allow portions of the requirement to be performed to ensure OPERABILITY of the batteries. These portions of the SR may only be performed in MODE 1, 2, 3, or 4 if the safety of the plant is maintained or enhanced. This change is acceptable because the relaxed Surveillance Requirement acceptance criteria are adequate to verify that the

equipment used to meet the LCO can perform its required functions. The change allows the performance of portions of the SR in MODE 1, 2, 3, or 4 provided plant safety is maintained or enhanced. Normally, the SR is conducted when the battery is not required to be OPERABLE. Under limited conditions, battery OPERABILITY may have to be demonstrated with the plant operating. Verification of battery OPERABILITY under these conditions is infrequent and the plant's safety is always maintained or enhanced by the performance of this requirement. The ITS requirement is consistent with the ISTS wording for this requirement. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.2.3 DOC LA01

LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action b. DOC L02 DOC M01 A. Required ~~One [or two]~~ battery charger~~[s]~~ on one subsystem inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	2 hours	4
	AND		
	A.2 Verify battery float current ≤ ~~[~~2~~]~~ amps.	Once per ~~[~~12~~]~~ hours	2
	AND		
	A.3 Restore battery charger~~[s]~~ to OPERABLE status.	~~[~~72~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~	
~~[B. One [or two] batter[y][ies on one subsystem] inoperable.~~	~~B.1 Restore batter[y][ies] to OPERABLE status.~~	~~[2] hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~	3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a B ~~C~~. One DC electrical power subsystem inoperable for reasons other than Condition A ~~[or B]~~.	B ~~C~~.1 Restore DC electrical power subsystem to OPERABLE status.	~~[~~2~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~
Action a. C ~~D~~. Required Action and Associated Completion Time not met.	C ~~D~~.1 Be in MODE 3. AND C ~~D~~.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.2.3.2.a.2 DOC LA02 SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.8.2.3.2.c.4	SR 3.8.4.2	Verify each battery charger supplies ≥ ~~[400]~~ 300 amps at ~~greater than or equal to the minimum established float voltage~~ 140 V for ≥ ~~[8]~~ 6 hours. OR Verify each battery charger can recharge the battery to the fully charged state within ~~[24]~~ 12 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ (1) (2)
4.8.2.3.2.e DOC M02 DOC L03	SR 3.8.4.3	----------NOTES---------- 1. The modified performance discharge test in SR 3.8.6.6 may be performed in lieu of SR 3.8.4.3. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ----------	
4.8.2.3.2.d		Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ (2)

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.2.1 DOC LA01

LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action b. DOC L02 DOC M01 A. ~~One [or two]~~ battery charger~~[s]~~ on one subsystem inoperable. (Required)	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage. AND A.2 Verify battery float current ≤ ~~[~~2~~]~~ amps. AND A.3 Restore battery charger~~[s]~~ to OPERABLE status.	2 hours (4) Once per ~~[~~12~~]~~ hours ~~[~~72~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ (2)
~~[B. One [or two] batter[y][ies on one subsystem] inoperable.~~	~~B.1 Restore batter[y][ies] to OPERABLE status.~~	~~[2] hours~~ ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]]~~ (3)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a. B. One DC electrical power subsystem inoperable for reasons other than Condition A ~~[or B]~~.	B. 1 Restore DC electrical power subsystem to OPERABLE status.	~~[~~2~~]~~ hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~
Action a. C. Required Action and Associated Completion Time not met.	C. 1 Be in MODE 3. AND C. 2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.2.1.a.2 DOC LA02 SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	~~[~~7 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
4.8.2.1.c.4	SR 3.8.4.2	Verify each battery charger supplies ≥ ~~[400]~~ 300 amps at ~~greater than or equal to the minimum established float voltage~~ 140 V for ≥ ~~[8]~~ 6 hours. OR Verify each battery charger can recharge the battery to the fully charged state within ~~[24]~~ 12 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2
DOC L02				
	SR 3.8.4.3	----------NOTES---------- 1. The modified performance discharge test in SR 3.8.6.6 may be performed in lieu of SR 3.8.4.3. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ----------		
4.8.2.1.e DOC M02				
DOC L03				
4.8.2.1.d		Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	~~[~~ ~~[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.4, DC SOURCES - OPERATING

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The bracketed ISTS 3.8.4 ACTION B has been deleted since it is not necessary. ISTS 3.8.4 ACTION C (ITS 3.8.4 ACTION B) covers the condition of an inoperable battery. Due to this deletion, the subsequent ACTIONS have been arranged in sequence.

4. Changed the word “One” to “Required” in ISTS 3.8.4 Condition A since there are two DC electrical power system chargers per DC electrical power subsystem and only one is required for each subsystem.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC ~~vital~~ bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of ~~Regulatory~~ Guide ~~1.~~6 (Ref. 2) and IEEE-308 (Ref. 3).

The ~~[~~125~~/250]~~ VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (~~[~~Train A and Train B~~]~~). Each subsystem consists of ~~[two]~~ 125 VDC batter~~ies~~ ~~[(each battery [50]% capacity)],~~ ~~the associated~~ battery charger~~(s)~~ for each battery, and all the associated control equipment and interconnecting cabling.

~~[The 250 VDC source is obtained by use of the two 125 VDC batteries connected in series. Additionally there is [one] spare battery~~ charger ~~per subsystem, which provides backup service in the event that the preferred battery charger is out of service~~. If the ~~spare~~ battery charger is substituted ~~for one of the preferred battery chargers, then~~ the requirements of independence and redundancy between subsystems are maintained.~~]~~

During normal operation, the ~~[~~125~~/250]~~ VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to ~~the battery charger,~~ the DC load is automatically powered from the station batteries.

The ~~[~~Train A and Train B~~]~~ DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, ~~[~~4.16~~]~~ kV switchgear, and ~~[~~480~~]~~ V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the AC ~~vital buses~~.

The DC power distribution system is described in more detail in the Bases for LCO 3.8.9, "Distributions System Operating," and for LCO 3.8.10, "Distribution Systems - Shutdown."

instrument
1
Safety
1
one
2
y, two redundant
1
(one required)
Each battery charger is individually adequate to support the train's required safety function. The redundant battery
The system has a third swing load group (AB) including a fifth battery charger on the AB bus that provides an alternate backup to either the two A-train or two B-train battery chargers.
2
AB
s operate continuously with load equally distributed by means of a load sharing device
one battery charger, the DC load is automatically powered from the one remaining battery charger. Upon loss of both normal battery chargers,
1
until the alternate backup charger can be aligned to the 125 VDC bus.
2
1
Instrument
System

BASES

BACKGROUND (continued)

Each 125~~/250~~ VDC battery is separately housed in a ventilated room ~~apart from its charger and distribution centers~~. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. ~~[~~There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.~~]~~

Each battery has adequate storage capacity to meet the duty cycle(s) discussed in the FSAR, Chapter ~~[~~8~~]~~ (Ref 4). The battery is designed with additional capacity above that required by the design duty cycle to allow for temperature variations and other factors.

The batteries for Train A and Train B DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit ~~is [105/210] V.~~

The battery cells are of flooded lead acid construction with a nominal specific gravity of ~~[~~1.215~~]~~. This specific gravity corresponds to an open circuit battery voltage of approximately ~~120~~ V for a ~~[58]~~ cell battery (i.e., cell voltage of ~~[2.065]~~ volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. ~~Once fully charged with its open circuit voltage ≥ [2.065] Vpc, the battery cell will maintain its capacity for [30] days without further charging per manufacturer's instructions.~~ Optimal long term performance however, is obtained by maintaining a float voltage ~~[~~2.20 to 2.25~~]~~ Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge. The nominal float voltage of ~~[2.22]~~ Vpc corresponds to a total float voltage output of ~~[128.8]~~ V for a ~~[58]~~ cell battery ~~as discussed in the FSAR, Chapter [8] (Ref. 4)~~.

Each Train A and Train B DC electrical power subsystem battery charger has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient excess capacity to restore the battery from the design minimum charge to its fully charged state within ~~[24]~~ hours while supplying normal steady state loads ~~discussed in the FSAR, Chapter [8] (Ref. 4)~~.

BASES

BACKGROUND (continued)

The battery charger is normally in the float-charge mode. Float-charge is the condition in which the charger is supplying the connected loads and the battery cells are receiving adequate current to optimally charge the battery. This assures the internal losses of a battery are overcome and the battery is maintained in a fully charged state.

When desired, the charger can be placed in the equalize mode. The equalize mode is at a higher voltage than the float mode and charging current is correspondingly higher. The battery charger is operated in the equalize mode after a battery discharge or for routine maintenance. Following a battery discharge, the battery recharge characteristic accepts current at the current limit of the battery charger (if the discharge was significant, e.g., following a battery service test) until the battery terminal voltage approaches the charger voltage setpoint. Charging current then reduces exponentially during the remainder of the recharge cycle. Lead-calcium batteries have recharge efficiencies of greater than 95%, so once at least 105% of the ampere-hours discharged have been returned, the battery capacity would be restored to the same condition as it was prior to the discharge. This can be monitored by direct observation of the exponentially decaying charging current or by evaluating the amp-hours discharged from the battery and amp-hours returned to the battery.

Battery parameters are described in further detail in the Bases for LCO 3.8.6, "Battery Parameters."

1

APPLICABLE SAFETY ANALYSES

U

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 5) and Chapter ~~[~~15~~]~~ (Ref. 6), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

2 1

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power and

b. A worst-case single failure.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The DC electrical power subsystems, each subsystem consisting of ~~[two]~~ batteries battery charger ~~[for each battery]~~ and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires ~~all required~~ batteries and ~~respective~~ chargers to be operating and connected to the associated DC bus(es).

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.5, "DC Sources - Shutdown."

ACTIONS

A.1, A.2, and A.3

Condition A represents one subsystem with one ~~[or two]~~ battery chargers inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within ~~[~~12~~]~~ hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

BASES

ACTIONS (continued)

~~REVIEWER'S NOTE~~

~~A plant that cannot meet the 12 hour Completion Time due to an inherent battery charging characteristic can propose an alternate time equal to 2 hours plus the time experienced to accomplish the exponential charging current portion of the battery charge profile following the service test (SR 3.8.4.3).~~ 3

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within ~~[~~12~~]~~ hours, avoiding a premature shutdown with its own attendant risk. 2

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 2 hours, and the charger is not operating in the current-limiting mode, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit mode that is necessary during the recovery period following a battery discharge event that the DC system is designed for.

If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within ~~[~~12~~]~~ hours (Required Action A.2). 2

Required Action A.2 requires that the battery float current be verified as less than or equal to ~~[~~2~~]~~ amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it is now fully capable of supplying the maximum expected load requirement. The ~~[~~2~~]~~ amp value is based on returning the battery to ~~[~~95~~]~~% charge and assumes a ~~[~~5~~]~~% design margin for the battery. If at the expiration of the initial ~~[~~12~~]~~ hour period the battery float current is not less than or equal to ~~[~~2~~]~~ amps this indicates there may be additional battery problems and the battery must be declared inoperable. 2

BASES

ACTIONS (continued)

~~REVIEWER'S NOTE~~
~~Any licensee wishing to adopt Completion Time greater than 72 hours for Required Action A.3 will need to demonstrate that the longer Completion Time is appropriate for the plant in accordance with the guidance in Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications", and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." Alternatively, a 7 day Completion Time can be justified by an acceptable method, such as a regulatory commitment that an alternate means to charge the batteries will be available that is capable of being supplied power from a power source that is independent of the offsite power supply. Otherwise, the 72 hour Completion Time must be adopted.~~

3

Required Action A.3 limits the restoration time for the inoperable battery charger to ~~[~~72~~]~~ hours. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used ~~(e.g., balance of plant non-Class 1E battery charger)~~. The ~~[~~72~~]~~ hour Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

2

~~B.1~~

5

~~REVIEWER'S NOTE'S~~
~~1. The 2 hour Completion Times of Required Actions B.1 and C.1 are in brackets. Any licensee wishing to request a longer Completion Time will need to demonstrate that the longer Completion Time is appropriate for the plant in accordance with the guidance in RG 1.177 and 1.174.~~

~~2. Condition B is included if Required Action B.1 (one [or two] batter[y][ies on one subsystem] inoperable) and Required Action C.1 (one DC electrical power subsystem inoperable for reasons other than Condition A [or B]) would have different Completion Times. If the plant design supports different Completion Times when a battery is inoperable but the charger is OPERABLE, then Condition B is used. If not, Condition B is deleted and only Condition C is used.~~

3

1

BASES

ACTIONS (continued)

~~Condition B represents one subsystem with one [or two] batter[y][ies] inoperable. With one [or two] batter[y][ies] inoperable, the DC bus is being supplied by the OPERABLE battery charger[s]. Any event that results in a loss of the AC bus supporting the battery charger[s] will also result in loss of DC to that subsystem. Recovery of the AC bus, especially if it is due to a loss of offsite power, will be hampered by the fact that many of the components necessary for the recovery (e.g., diesel generator control and field flash, AC load shed and diesel generator output circuit breakers, etc.) likely rely upon the batter[y][ies]. In addition the energization transients of any DC loads that are beyond the capability of the battery charger[s] and normally require the assistance of the batter[y][ies] will not be able to be brought online. The [2] hour limit allows sufficient time to effect restoration of an inoperable battery given that the majority of the conditions that lead to battery inoperability (e.g., loss of battery charger, battery cell voltage less than [2.07] V, etc.) are identified in Specifications 3.8.4, 3.8.5, and 3.8.6 together with additional specific Completion Times. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 2

~~C~~.1

B

Condition ~~C~~ represents one subsystem with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected subsystem. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution subsystem. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 5 2

If one of the required DC electrical power subsystems is inoperable for reasons other than Condition A ~~or B~~ (e.g., inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure could, however, result in the loss of the minimum necessary DC electrical subsystems to mitigate a worst case accident, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 7) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown. 5

BASES

ACTIONS (continued)

~~D~~.1 and ~~D~~.2

C

5

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

3

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

C

Required Action ~~D~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

5

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

topical report, CE NPSD-1186-A

The Completion Time to bring the unit to MODE ~~5~~ 4 is consistent with ~~the time required in Regulatory Guide 1.93~~ (Ref. ~~7~~ 8). 5 1

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the battery chargers, which support the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the minimum float voltage established by the battery manufacturer (~~[~~2.20~~]~~ Vpc times the number of connected cells or ~~[127.6]~~ 132 V for a ~~58~~ 60 cell battery at the battery terminals). This voltage maintains the battery plates in a condition that supports maintaining the grid life. ~~[The 7 day Frequency is consistent with manufacturer recommendations.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 1 2

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 3 2

SR 3.8.4.2

Safety

This SR verifies the design capacity of the battery chargers. According to ~~Regulatory~~ Guide ~~1.~~32 (Ref. 9), the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

300
140 V
6

This SR provides two options. One option requires that each battery charger be capable of supplying ~~[400]~~ amps at ~~the minimum established float voltage~~ for ~~[8]~~ hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power. The time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least ~~[~~2~~]~~ hours. 1 2

The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, and the exponential decay in charging current. The battery is recharged when the measured charging current is ≤ ~~[~~2~~]~~ amps.

~~[The Surveillance Frequency of 18 months is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these [18 month] intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 3 2

SR 3.8.4.3

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[The Surveillance Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed during refueling operations, or at some other outage, with intervals between tests not to exceed [18 months].~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
]~~

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 ~~or 2~~ is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 ~~or~~2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. ~~Re~~gulatory Guide ~~1.~~6, March 10, 1971.
3. IEEE-308-~~[~~197~~8]~~.

BASES

REFERENCES (continued)

4. FSAR, Chapter [8].

5. FSAR, Chapter [6].

6. FSAR, Chapter [15].

7. Regulatory Guide 1.93, December 1974.

8. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

9. ~~Regulatory~~ Guide ~~1.~~32~~, February 1977~~.

~~10. Regulatory Guide 1.129, December 1974.~~

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC ~~vital~~ bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The ~~[~~125~~/250]~~ VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (~~[~~Train A and Train B~~]~~). Each subsystem consists of ~~[two]~~ 125 VDC batter~~ies~~ ~~[(each battery [50]% capacity)],~~ ~~the associated~~ battery charger~~(s)~~ for each battery, and all the associated control equipment and interconnecting cabling.

~~[The 250 VDC source is obtained by use of the two 125 VDC batteries connected in series. Additionally there is [one] spare battery~~ charger ~~per subsystem, which provides backup service in the event that the preferred battery charger is out of service~~. If the ~~spare~~ battery charger is substituted ~~for one of the preferred battery chargers, then~~ the requirements of independence and redundancy between subsystems are maintained.~~]~~

During normal operation, the ~~[~~125~~/250]~~ VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to ~~the battery charger,~~ the DC load is automatically powered from the station batteries.

The ~~[~~Train A and Train B~~]~~ DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, ~~[~~4.16~~]~~ kV switchgear, and ~~[~~480~~]~~ V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the AC ~~vital buses~~.

The DC power distribution system is described in more detail in the Bases for LCO 3.8.9, "Distributions System Operating," and for LCO 3.8.10, "Distribution Systems - Shutdown."

instrument
1
1
2
one
y, two redundant
(one required)
1
Each battery charger is individually adequate to support the train's required safety function. The redundant battery
The system has a third swing load group (AB) including a fifth battery charger on the AB bus that provides an alternate backup to either the two A-train or two B-train battery chargers .
2
AB
s operate continuously with load equally distributed by means of a load sharing device
one battery charger, the DC load is automatically powered from the one remaining battery charger. Upon loss of both normal battery chargers,
1
until the alternate backup charger can be aligned to the 125 VDC bus.
2
1
Instrument
System

BASES

BACKGROUND (continued)

Each 125~~/250~~ VDC battery is separately housed in a ventilated room ~~apart from its charger and distribution centers~~. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. ~~[~~There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.~~]~~

Each battery has adequate storage capacity to meet the duty cycle(s) discussed in the FSAR, Chapter ~~[~~8~~]~~ (Ref 4). The battery is designed with additional capacity above that required by the design duty cycle to allow for temperature variations and other factors.

The batteries for Train A and Train B DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit ~~is [105/210] V.~~

of 105 V is bounded by the minimum calculated DC System voltage required to support the DC loads.

The battery cells are of flooded lead acid construction with a nominal specific gravity of ~~[~~1.215~~]~~. This specific gravity corresponds to an open circuit battery voltage of approximately ~~[~~120 V for a ~~[~~58~~]~~ cell battery (i.e., cell voltage of ~~[~~2.065~~]~~ volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. ~~Once fully charged with its open circuit voltage ≥ [2.065] Vpc, the battery cell will maintain its capacity for [30] days without further charging per manufacturer's instructions.~~ Optimal long term performance however, is obtained by maintaining a float voltage ~~[~~2.20 to 2.25~~]~~ Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge. The nominal float voltage of ~~[2.22]~~ Vpc corresponds to a total float voltage output of ~~[128.8]~~ V for a ~~[58]~~ cell battery ~~as discussed in the FSAR, Chapter [8] (Ref. 4)~~.

Each Train A and Train B DC electrical power subsystem battery charger has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient excess capacity to restore the battery from the design minimum charge to its fully charged state within ~~[~~24~~]~~ hours while supplying normal steady state loads ~~discussed in the FSAR, Chapter [8] (Ref. 4)~~.

BASES

BACKGROUND (continued)

The battery charger is normally in the float-charge mode. Float-charge is the condition in which the charger is supplying the connected loads and the battery cells are receiving adequate current to optimally charge the battery. This assures the internal losses of a battery are overcome and the battery is maintained in a fully charged state.

When desired, the charger can be placed in the equalize mode. The equalize mode is at a higher voltage than the float mode and charging current is correspondingly higher. The battery charger is operated in the equalize mode after a battery discharge or for routine maintenance. Following a battery discharge, the battery recharge characteristic accepts current at the current limit of the battery charger (if the discharge was significant, e.g., following a battery service test) until the battery terminal voltage approaches the charger voltage setpoint. Charging current then reduces exponentially during the remainder of the recharge cycle. Lead-calcium batteries have recharge efficiencies of greater than 95%, so once at least 105% of the ampere-hours discharged have been returned, the battery capacity would be restored to the same condition as it was prior to the discharge. This can be monitored by direct observation of the exponentially decaying charging current or by evaluating the amp-hours discharged from the battery and amp-hours returned to the battery.

Battery parameters are described in further detail in the Bases for LCO 3.8.6, "Battery Parameters."

1

APPLICABLE SAFETY ANALYSES

U

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 5) and Chapter ~~[~~15~~]~~ (Ref. 6), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

2 1

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power and

b. A worst-case single failure.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The DC electrical power subsystems, each subsystem consisting of ~~[two]~~ batteries battery charger ~~[for each battery]~~ and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires ~~all required~~ batteries and ~~respective~~ chargers to be operating and connected to the associated DC bus(es).

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.5, "DC Sources - Shutdown."

ACTIONS

A.1, A.2, and A.3

Condition A represents one subsystem with one ~~[or two]~~ battery chargers inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within ~~[~~12~~]~~ hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

BASES

ACTIONS (continued)

~~REVIEWER'S NOTE~~

~~A plant that cannot meet the 12 hour Completion Time due to an inherent battery charging characteristic can propose an alternate time equal to 2 hours plus the time experienced to accomplish the exponential charging current portion of the battery charge profile following the service test (SR 3.8.4.3).~~ 3

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within ~~[~~12~~]~~ hours, avoiding a premature shutdown with its own attendant risk. 2

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 2 hours, and the charger is not operating in the current-limiting mode, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit mode that is necessary during the recovery period following a battery discharge event that the DC system is designed for.

If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within ~~[~~12~~]~~ hours (Required Action A.2). 2

Required Action A.2 requires that the battery float current be verified as less than or equal to ~~[~~2~~]~~ amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it is now fully capable of supplying the maximum expected load requirement. The ~~[~~2~~]~~ amp value is based on returning the battery to ~~[~~95~~]~~% charge and assumes a ~~[~~5~~]~~% design margin for the battery. If at the expiration of the initial ~~[~~12~~]~~ hour period the battery float current is not less than or equal to ~~[~~2~~]~~ amps this indicates there may be additional battery problems and the battery must be declared inoperable. 2

BASES

ACTIONS (continued)

~~REVIEWER'S NOTE~~

~~Any licensee wishing to adopt Completion Time greater than 72 hours for Required Action A.3 will need to demonstrate that the longer Completion Time is appropriate for the plant in accordance with the guidance in Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications", and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." Alternatively, a 7 day Completion Time can be justified by an acceptable method, such as a regulatory commitment that an alternate means to charge the batteries will be available that is capable of being supplied power from a power source that is independent of the offsite power supply. Otherwise, the 72 hour Completion Time must be adopted.~~

3

Required Action A.3 limits the restoration time for the inoperable battery charger to ~~[~~72~~]~~ hours. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used ~~(e.g., balance of plant non-Class 1E battery charger)~~. The ~~[~~72~~]~~ hour Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

2

~~B.1~~

5

~~REVIEWER'S NOTE'S~~

~~1. The 2 hour Completion Times of Required Actions B.1 and C.1 are in brackets. Any licensee wishing to request a longer Completion Time will need to demonstrate that the longer Completion Time is appropriate for the plant in accordance with the guidance in RG 1.177 and 1.174.~~

~~2. Condition B is included if Required Action B.1 (one [or two] batter[y][ies on one subsystem] inoperable) and Required Action C.1 (one DC electrical power subsystem inoperable for reasons other than Condition A [or B]) would have different Completion Times. If the plant design supports different Completion Times when a battery is inoperable but the charger is OPERABLE, then Condition B is used. If not, Condition B is deleted and only Condition C is used.~~

3

BASES

ACTIONS (continued)

~~Condition B represents one subsystem with one [or two] batter[y][ies] inoperable. With one [or two] batter[y][ies] inoperable, the DC bus is being supplied by the OPERABLE battery charger[s]. Any event that results in a loss of the AC bus supporting the battery charger[s] will also result in loss of DC to that subsystem. Recovery of the AC bus, especially if it is due to a loss of offsite power, will be hampered by the fact that many of the components necessary for the recovery (e.g., diesel generator control and field flash, AC load shed and diesel generator output circuit breakers, etc.) likely rely upon the batter[y][ies]. In addition the energization transients of any DC loads that are beyond the capability of the battery charger[s] and normally require the assistance of the batter[y][ies] will not be able to be brought online. The [2] hour limit allows sufficient time to effect restoration of an inoperable battery given that the majority of the conditions that lead to battery inoperability (e.g., loss of battery charger, battery cell voltage less than [2.07] V, etc.) are identified in Specifications 3.8.4, 3.8.5, and 3.8.6 together with additional specific Completion Times. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 2

~~C~~.1

B

Condition ~~C~~ represents one subsystem with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected subsystem. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution subsystem. ~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 5 2

If one of the required DC electrical power subsystems is inoperable for reasons other than Condition A ~~or B~~ (e.g., inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure could, however, result in the loss of the minimum necessary DC electrical subsystems to mitigate a worst case accident, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 7) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown. 5

BASES

ACTIONS (continued)

~~D~~.1 and ~~D~~.2

C

5

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

3

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

C

Required Action ~~D~~.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

5

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

The Completion Time to bring the unit to MODE ~~5~~ is consistent with ~~the time required in Regulatory Guide 1.93~~ (Ref. ~~7~~).

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the battery chargers, which support the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the minimum float voltage established by the battery manufacturer (~~[~~2.20~~]~~ Vpc times the number of connected cells or ~~[127.6]~~ V for a ~~58~~ cell battery at the battery terminals). This voltage maintains the battery plates in a condition that supports maintaining the grid life. ~~[The 7 day Frequency is consistent with manufacturer recommendations.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

SR 3.8.4.2

This SR verifies the design capacity of the battery chargers. According to Regulatory Guide 1.32 (Ref. 9), the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR provides two options. One option requires that each battery charger be capable of supplying ~~[400]~~ 300 amps at ~~the minimum established float voltage~~ 140 V for ~~[8]~~ 6 hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power. The time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least ~~[~~2~~]~~ hours. 1 2

The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, and the exponential decay in charging current. The battery is recharged when the measured charging current is ≤ ~~[~~2~~]~~ amps.

~~[The Surveillance Frequency of 18 months is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these [18 month] intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------~~ 3
~~]~~ 2

SR 3.8.4.3

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[The Surveillance Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed during refueling operations, or at some other outage, with intervals between tests not to exceed [18 months].~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 ~~or 2~~ is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 ~~or 2~~. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

, 2, 3, or 4

, 2, 3, or 4

1.6ERENCES

1. 10 CFR 50, Appendix A, GDC 17. (as amended through February 20, 1971)
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-~~[~~197~~8]~~.

BASES

REFERENCES (continued)

4. FSAR, Chapter [8].

5. FSAR, Chapter [6].

6. FSAR, Chapter [15].

7. Regulatory Guide 1.93, December 1974.

8. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

9. Regulatory Guide 1.32, February 1977.

10. Regulatory Guide 1.129, December 1974.

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.4 BASES, DC SOURCES - OPERATING

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.
3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.
4. These battery design values have been deleted because they are more specific than necessary and are not required to provide sufficient background for this Specification.
5. Changes have been made to be consistent with changes made to the Specifications.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.8.4, DC SOURCES - OPERATING

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

ITS 3.8.5, DC Sources - Shutdown

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ELECTRICAL POWER SYSTEMS

SOURCES

D.C. ~~DISTRIBUTION~~ - SHUTDOWN

LIMITING CONDITION FOR OPERATION

See ITS 3.8.10

LA01

One

power subsystem

LCO 3.8.5 3.8.2.4 ~~As a minimum, the following~~ D.C. electrical ~~equipment and bus~~ shall be energized ~~and~~ OPERABLE~~:~~

1 - 125-volt D.C. bus, and — See ITS 3.8.10

~~1 - 125-volt battery bank and charger supplying the above D.C. bus.~~

LA01

Applicability **APPLICABILITY**: MODES 5 and 6~~:~~ ,

During movement of irradiated fuel assemblies.

ACTION **ACTION**:

M01

Add proposed ACTIONS NOTE

~~With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.~~ — Add proposed ACTION A

L01

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying indicated power availability.

See ITS 3.8.10

Add proposed SR 3.8.5.1 NOTE

L02

SR 3.8.5.1 4.8.2.4.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

LA01

DC electrical power subsystem

LCO 3.8.5 3.8.2.2 ~~As a minimum,~~ one ~~125-volt battery bank and a full capacity charger~~ shall be OPERABLE.

Applicability **APPLICABILITY**: MODES 5 and 6~~:~~ ~~,~~

During movement of irradiated fuel assemblies.

M01

ACTION **ACTION**:

Add proposed ACTIONS NOTE

ACTION A.1, A.2, A.3

a. With the required battery bank inoperable, immediately suspend ~~all operations involving CORE ALTERATIONS,~~ operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, or movement of irradiated fuel; initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible~~, and within 8 hours, depressurize and vent the Reactor Coolant System through a 3.58 square inch vent~~.

L05 R1

L03

b. ~~With the required full capacity charger inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.~~

L04

SURVEILLANCE REQUIREMENTS

L02

Add proposed SR 3.8.5.1 NOTE

SR 3.8.5.1 4.8.2.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

DISCUSSION OF CHANGES
ITS 3.8.5, DC SOURCES - SHUTDOWN

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.8.2.4 and Unit 2 CTS 3.8.2.2 are applicable in MODES 5 and 6. ITS 3.8.5 is applicable in MODES 5 and 6 and during movement of irradiated fuel assemblies. A Note has been added to the ACTIONS which states that LCO 3.0.3 is not applicable. This changes the CTS by adding the Applicability of during movement of irradiated fuel assemblies and adds the Note to the ACTIONS stating that LCO 3.0.3 is not applicable.

This change is acceptable because the proposed requirements are necessary to ensure the DC electrical power sources are OPERABLE to support equipment required to be OPERABLE during movement of irradiated fuel assemblies. Movement of fuel normally occurs during MODES 5 and 6, however, it can also occur outside of containment in other plant MODES (MODES 1, 2, 3, and 4) or other conditions (i.e., reactor defueled). This Specification is needed to ensure the appropriate DC electrical power source requirements are specified during fuel handling and ensure the appropriate actions are taken (i.e., stop fuel movement) when the minimum electrical supplies are not available (See Unit 1 DOC L01 and Unit 2 DOCs L03 and L04 for the changes to the Required Actions). This change adds a clarification Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODES 5 or 6, LCO 3.0.3 is not applicable and would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations and the inability to suspend movement in accordance with the ITS 3.8.5 Required Actions would not be sufficient reason to require a reactor shutdown. This Note has been added for clarification and is necessary since defaulting to LCO 3.0.3 would require the reactor to be shutdown but would not require suspension of activities with a potential for releasing radioactive materials. This change is designated as more restrictive because the ITS requires the equipment to be OPERABLE during movement of irradiated fuel assemblies both inside and outside of the containment in conditions other than MODES 5 and 6.

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS 3.8.5, DC SOURCES - SHUTDOWN

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.8.2.4 states, in part, 1 - 125-volt battery bank and charger supplying the DC bus. Unit 2 CTS 3.8.2.2 requires a minimum of one 125 volt battery and a full capacity charger. ITS LCO 3.8.5 requires one DC electrical power subsystem to be OPERABLE. This changes the CTS by moving the details of the DC electrical power subsystems from the CTS to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the OPERABILITY requirements for electrical power trains. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Control Program in Chapter 5. The program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 **Unit 1 only:** *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.8.2.4 ACTION requires the establishment of containment integrity within 8 hours with less than the minimum complement of DC sources OPERABLE. ITS 3.8.5 ACTION A requires suspending movement of irradiated fuel assemblies, suspending operations involving a positive reactivity addition that could result in the loss of required SDM or boron concentration, and the initiation of action to restore required DC electrical power sources to OPERABLE status. This changes the CTS by replacing the existing Required Action to restore containment integrity with Actions that will prevent a fuel handling accident from occurring and Actions that will ensure the reactor remains subcritical.

The purpose of the CTS 3.8.2.4 Action is to isolate the containment to minimize any release from the plant if an event were to occur during shutdown conditions with no DC Sources OPERABLE. This change is acceptable because the ITS Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The proposed Required Actions require the suspension of movement of irradiated fuel assemblies, suspension of operations involving positive reactivity additions that could result in the loss of required SDM or boron concentration, and the initiation of actions to restore required DC electrical power subsystem to OPERABLE status. Suspending the movement of irradiated fuel assemblies will prevent a fuel handling accident from occurring and suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit will ensure the reactor remains subcritical. The requirement to initiate action to restore the required DC electrical power subsystem to OPERABLE status will

ensure the plant is placed in compliance with the LCO in an expeditious manner. The proposed actions will also minimize the potential for any accident releases outside of the containment and are considered acceptable in lieu of the current action to restore containment integrity within 8 hours. The actions may be considered somewhat more restrictive since immediate action is required, however, is classified as less restrictive since the current actions to restore containment integrity have been deleted. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 7 - Relaxation Of Surveillance Frequency)* Unit 1 CTS 4.8.2.4.2 and Unit 2 CTS 4.8.2.2 require the demonstration of the OPERABILITY of the 125 VDC battery banks and chargers in accordance with the Surveillance Requirements of Unit 1 CTS 4.8.2.3.2 and Unit 2 CTS 4.8.2.1. ITS SR 3.8.5.1 requires SR 3.8.4.1, SR 3.8.4.2, and SR 3.8.4.3 to be applicable. However, a Note has been added that states ITS SR 3.8.4.2 and SR 3.8.4.3 do not have to be performed. This changes the CTS by allowing certain SRs not to be performed.

The purpose of the ITS SR 3.8.5.1 Note is to ensure that required equipment is not made inoperable by testing when the equipment is the only OPERABLE equipment available to support unit operations. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. ITS SR 3.8.4.3 is the battery capacity test. ITS SR 3.8.4.2 is a battery charger capacity test. The performance of these SRs involves tests that would cause the only required OPERABLE DC electrical power subsystem components to be rendered inoperable. This condition presents a significant risk if an event were to occur during test performance. In an effort to address this concern, ITS SR 3.8.5.1 has a Note that excludes performance requirements of any Surveillance that would require the required OPERABLE DC electrical power subsystem components to be rendered inoperable. This allowance does not take exception to the requirement for the DC electrical power subsystem to be capable of performing the particular function, but just to the requirement to demonstrate that capability while the required DC source of power is being relied on to support meeting the LCO. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L03 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* Unit 2 CTS 3.8.2.2 Action b states, in part, “with the required battery bank inoperable, within 8 hours, depressurize and vent the Reactor Coolant System through a 3.58 square inch vent. ITS 3.8.5 does not include a required action for depressurizing and venting the Reactor Coolant System. This changes the CTS by removing the requirement to depressurize and vent the Reactor Coolant System within 8 hours with the required battery bank inoperable.

The purpose of minimum DC electrical power subsystem technical specification required actions is to establish mitigating actions with appropriate time limitations to restore the minimum required DC electrical power subsystem. This change is acceptable because CTS 3.4.9.3 provides the operability requirements,

mitigating actions, and time limitations for low temperature overpressure protection (LTOP) systems. CTS 3.4.9.3 Action a.1 requires depressurization and venting of the Reactor Coolant System when the LTOP System is inoperable for any reason including the loss of DC power to required LTOP components. ITS 3.4.12 retains the required CTS ACTION. Refer to ITS 3.4.12 Discussion of Changes for changes to CTS 3.4.9.3 actions. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L04 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* Unit 2 CTS 3.8.2.2 Action b in part states with the required full capacity charger inoperable, demonstrate the OPERABILITY of its associated battery bank. ITS LCO 3.8.5 does not include Actions for an inoperable required battery charger for plants that require only one DC electrical power subsystem to be OPERABLE during shutdown conditions. This changes CTS by removing the CTS ACTIONS associated with a required full capacity charger inoperable.

The purpose of the ITS 3.8.5 Required Actions is to allow appropriate remedial actions with appropriate time limitations to restore a required inoperable DC electrical power subsystem to OPERABLE status. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk while providing time to restore inoperable features. This change establishes the appropriate actions to be taken with an inoperable required DC electrical power subsystem for the PSL Unit 2 design. The proposed Required Actions require the suspension of movement of irradiated fuel assemblies, suspension of operations involving positive reactivity additions that could result in the loss of required SDM or boron concentration, and the initiation of actions to restore required DC electrical power subsystem to OPERABLE status. Suspending the movement of irradiated fuel assemblies will prevent a fuel handling accident from occurring and suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit will ensure the reactor remains subcritical. The requirement to initiate action to restore the required DC electrical power subsystem to OPERABLE status will ensure the plant is placed in compliance with the LCO in an expeditious manner. The ITS requirement is consistent with the ISTS wording for this requirement. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L05 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* Unit 2 CTS 3.8.2.2 ACTION states, in part, “With the required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS,...” ITS 3.8.5 ACTIONS do not include the requirement to suspend CORE ALTERATIONS. This changes the CTS by removing the Required Action to suspend all operations involving CORE ALTERATIONS. R1

The purpose of the CTS 3.8.2.2 ACTION is to minimize the possibility of an event that may require the electrical power source to mitigate the consequences of a design basis event. CORE ALTERATIONS is defined in CTS 1.9, in part, as "the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity, within the reactor vessel with the head

removed and fuel in the vessel." Core Alterations only occur when the reactor vessel head is removed - it only applies in MODE 6. There is only one accident considered during MODE 6 that involves a Core Alteration - a fuel handling accident. According to UFSAR Section 15.7.4.1.2, a fuel handling accident is initiated by the dropping of an irradiated fuel assembly either in the containment or in the fuel handling building. Suspension of CORE ALTERATIONS, except for suspension of movement of irradiated fuel, will not prevent or impair the mitigation of a fuel handling accident. ITS 3.8.5 retains the requirement to suspend movement of irradiated fuel assemblies as Required Action A.1 (for an inoperable DC electrical power subsystem). R1

Deleting the reference to “CORE ALTERATIONS” is acceptable because the only CORE ALTERATION analyzed in the safety analysis that is potentially affected by an inoperable DC electrical power subsystem is covered by the ITS Required Actions. Additionally, CORE ALTERATIONS that may be capable of damaging a fuel assembly prior to the 72 hour decay period is improbable because of the physical time required to perform plant shutdown, cooldown, depressurize the Reactor Coolant System (RCS), and the additional operations required to expose irradiated fuel in the reactor vessel (e.g., containment entry, removal of vessel head, removal of vessel internals, etc.) is greater than 72 hours. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

3.8.2.4 LCO 3.8.5

~~[DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown."]~~ 1

~~[~~One DC electrical power subsystem shall be OPERABLE.~~]~~ 3

~~----------REVIEWER'S NOTE----------~~
~~This second option above applies for plants having a pre-ITS licensing basis (CTS) for electrical power requirements during shutdown conditions that required only one DC electrical power subsystem to be OPERABLE. Action A and the bracketed optional wording in Condition B are also eliminated for this case. The first option above is adopted for plants that have a CTS requiring the same level of DC electrical power subsystem support as is required for power operating conditions.~~
~~----------~~ 1

Applicability DOC M01 APPLICABILITY: MODES 5 and 6,
During movement of ~~[recently]~~ irradiated fuel assemblies. 3

ACTIONS

----------NOTE----------
DOC M01 LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~[A. One [or two] battery charger[s on one subsystem] inoperable.~~	~~A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.~~	~~2 hours~~
~~AND~~	~~AND~~	1
~~The redundant subsystem battery and charger[s] OPERABLE.~~	~~A.2 Verify battery float current ≤ [2] amps.~~	~~Once per [12] hours~~
	~~AND~~	

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
	~~A.3 Restore battery charger[s] to OPERABLE status.~~	~~[72] hours~~	1
3.8.2.4 Action DOC L01 A R ~~B~~. ~~One [or more]~~ required DC electrical power subsystem~~[s]~~ inoperable ~~[for reasons other than Condition A.~~ ~~OR~~ ~~Required Action and associated Completion Time of Condition A not met~~].	~~B.1 Declare affected required feature(s) inoperable.~~ ~~OR~~ A.1 ~~B.2.1~~ Suspend movement of ~~[recently]~~ irradiated fuel assemblies. ← AND A.2 ~~B.2.2~~ Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. ← AND A.3 ~~B.2.3~~ Initiate action to restore required DC electrical power subsystems to OPERABLE status.	~~Immediately~~ Immediately Immediately Immediately	1 2 1 3 1 1

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.2.4.2 DOC L02 SR 3.8.5.1	----------NOTE---------- The following SRs are not required to be performed: SR 3.8.4.2 and SR 3.8.4.3. ----------	
	the For DC source~~s~~ required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.2 SR 3.8.4.3	In accordance with applicable SRs (2)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

3.8.2.2 LCO 3.8.5

~~[DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown."]~~ 1

~~[~~One DC electrical power subsystem shall be OPERABLE.~~]~~ 3

~~----------REVIEWER'S NOTE----------~~
~~This second option above applies for plants having a pre-ITS licensing basis (CTS) for electrical power requirements during shutdown conditions that required only one DC electrical power subsystem to be OPERABLE. Action A and the bracketed optional wording in Condition B are also eliminated for this case. The first option above is adopted for plants that have a CTS requiring the same level of DC electrical power subsystem support as is required for power operating conditions.~~
~~----------~~ 1

Applicability
DOC M01

APPLICABILITY: MODES 5 and 6,
During movement of ~~[recently]~~ irradiated fuel assemblies. 3

ACTIONS

----------NOTE----------
DOC M01 LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~[A. One [or two] battery charger[s on one subsystem] inoperable.~~	~~A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.~~	~~2 hours~~
~~AND~~	~~AND~~	1
~~The redundant subsystem battery and charger[s] OPERABLE.~~	~~A.2 Verify battery float current ≤ [2] amps.~~	~~Once per [12] hours~~
	~~AND~~	

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
	~~A.3 Restore battery charger[s] to OPERABLE status.~~	~~[72] hours~~	1
3.8.2.4 Action ~~B~~. ~~One [or more]~~ required DC electrical power subsystem~~[s]~~ inoperable ~~[for reasons other than Condition A.~~ ~~OR~~ ~~Required Action and associated Completion Time of Condition A not met]~~.	~~B.1 Declare affected required feature(s) inoperable.~~ ~~OR~~ ~~B.2.1~~ Suspend movement of ~~[recently]~~ irradiated fuel assemblies. AND ~~B.2.2~~ Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. AND ~~B.2.3~~ Initiate action to restore required DC electrical power subsystems to OPERABLE status.	~~Immediately~~ Immediately Immediately Immediately	2 1 1 3 1 1

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1 (4.8.2.2 DOC L02)	----------NOTE---------- The following SRs are not required to be performed: SR 3.8.4.2 and SR 3.8.4.3. ----------	
	For the DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.2 SR 3.8.4.3	In accordance with applicable SRs (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.5, DC SOURCES - SHUTDOWN

1. The bracketed optional ISTS LCO 3.8.5 and "Reviewers Note" have been deleted since the current licensing basis only requires one DC electrical power subsystem to be OPERABLE. ISTS 3.8.5 ACTION A has been deleted since only one required DC electrical power train is specified in the LCO. This allowance is only acceptable if the first option of the LCO is used. The subsequent Condition and Required Actions have been renumbered and modified, as applicable.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."

APPLICABLE SAFETY ANALYSES

U

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 1) and Chapter ~~[~~15~~]~~ (Ref. 2), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation. 1 2

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of ~~[recently]~~ irradiated fuel assemblies ensures that: 2

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident ~~[involving handling recently irradiated fuel~~. ~~Due to radioactive decay, DC electrical power is only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)].~~ 2

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many DBAs that are

BASES

APPLICABLE SAFETY ANALYSES (continued)

analyzed in MODES [1, 2, 3, and 4] have no specific analyses in MODES [5 and 6] because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems. 2

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The DC electrical power subsystem[s, [each required] [the required] subsystem] consisting of [two] batter[ies], one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the subsystem, [are] [is] required to be OPERABLE to support [required] [one] subsystem[s] of distribution systems required [OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown."] This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents [involving handling recently irradiated fuel]).

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required
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BASES

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6, and during movement of ~~[recently]~~ irradiated fuel assemblies provide assurance that: (2)

a. Required features needed to mitigate a fuel handling ~~[involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~ accident are available, (2)

b. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

~~A.1, A.2, and A.3~~ (3)

~~REVIEWER'S NOTE~~

~~ACTION A is included only when plant-specific implementation of LCO 3.8.5 includes the potential to require both subsystems of the DC System to be OPERABLE. If plant-specific implementation results in LCO 3.8.5 requiring only one subsystem of the DC System to be OPERABLE, then ACTION A is omitted and ACTION B is renumbered as ACTION A.~~ (4)

~~Condition A represents one subsystem with one [or two] battery chargers inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or~~ (3)

BASES

ACTIONS (continued)

equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within [12] hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

---REVIEWER'S NOTE---
A plant that cannot meet the 12 hour Completion Time due to an inherent battery charging characteristic can propose an alternate time equal to 2 hours plus the time experienced to accomplish the exponential charging current portion of the battery charge profile following the service test (SR 3.8.4.3).

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within [12] hours, avoiding a premature shutdown with its own attendant risk. 3

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 2 hours, and the charger is not operating in the current-limiting modes, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit modes that is necessary during the recovery period following a battery discharge event that the DC system is designed for.

If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within [12] hours (Required Action A.2).

BASES

ACTIONS (continued)

~~Required Action A.2 requires that the battery float current be verified as less than or equal to [2] amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial [12] hour period the battery float current is not less than or equal to [2] amps this indicates there may be additional battery problems and the battery must be declared inoperable.~~

~~----------REVIEWER'S NOTE----------~~
~~Any licensee wishing to adopt a Completion Time greater than 72 hours for Required Action A.3 will need to demonstrate that the Completion Time is appropriate for the plant in accordance with the guidance in Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications." Otherwise, the 72 hour Completion Time must be adopted.~~
~~----------~~

~~Required Action A.3 limits the restoration time for the inoperable battery charger to [72] hours. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., balance of plant non-Class 1E battery charger). The [72] hour Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.~~

3

A.1, A.2, and A.3

~~B.1, B.2.1, B.2.2, and B.2.3~~

With the required DC electrical power subsystem inoperable, the minimum required DC electrical power source is not available. Therefore, suspension of

3

~~[If two subsystems are required by LCO 3.8.10, the remaining subsystem with DC power available may be capable of supporting sufficient systems to allow continuation of [recently] irradiated fuel movement]. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend~~ movement of ~~[recently]~~ irradiated fuel assemblies, and operations involving positive reactivity additions~~)~~ that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)~~.~~ Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safety operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than ~~that~~ what

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is required

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St. Lucie – Unit 1

1

BASES

ACTIONS (continued)

would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystem~~[s]~~ and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems. (2)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystem~~s~~ should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power. (1)

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 states that Surveillances required by SR 3.8.4.1 through SR 3.8.4.3 are applicable in these MODES. See the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. FSAR, Chapter ~~[~~6~~]~~.
2. FSAR, Chapter ~~[~~15~~]~~.

(U) (2) (1)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."

APPLICABLE SAFETY ANALYSES

U

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 1) and Chapter ~~[~~15~~]~~ (Ref. 2), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation. 1 2

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of ~~[recently]~~ irradiated fuel assemblies ensures that: 2

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident ~~[involving handling recently irradiated fuel~~. ~~Due to radioactive decay, DC electrical power is only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)].~~ 2

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many DBAs that are

BASES

APPLICABLE SAFETY ANALYSES (continued)

analyzed in MODES [1, 2, 3, and 4] have no specific analyses in MODES [5 and 6] because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems. 2

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The DC electrical power subsystem[s, [each required] [the required] subsystem] consisting of [two] batter[ies], one battery charger per battery, and the corresponding control equipment and interconnecting cabling within the subsystem, [are] [is] required to be OPERABLE to support [required] [one] subsystem[s] of distribution systems required [OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown."] This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents [involving handling recently irradiated fuel]).

one
y
required
2
1
2

BASES

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6, and during movement of ~~[recently]~~ irradiated fuel assemblies provide assurance that: (2)

a. Required features needed to mitigate a fuel handling ~~[involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~ accident are available, (2)

b. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

~~A.1, A.2, and A.3~~ (3)

~~REVIEWER'S NOTE~~

~~ACTION A is included only when plant-specific implementation of LCO 3.8.5 includes the potential to require both subsystems of the DC System to be OPERABLE. If plant-specific implementation results in LCO 3.8.5 requiring only one subsystem of the DC System to be OPERABLE, then ACTION A is omitted and ACTION B is renumbered as ACTION A.~~ (4)

~~Condition A represents one subsystem with one [or two] battery chargers inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or~~ (3)

BASES

ACTIONS (continued)

~~equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within [12] hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.~~

~~REVIEWER'S NOTE~~
~~A plant that cannot meet the 12 hour Completion Time due to an inherent battery charging characteristic can propose an alternate time equal to 2 hours plus the time experienced to accomplish the exponential charging current portion of the battery charge profile following the service test (SR 3.8.4.3).~~

~~A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within [12] hours, avoiding a premature shutdown with its own attendant risk.~~ 3

~~If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 2 hours, and the charger is not operating in the current-limiting modes, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit modes that is necessary during the recovery period following a battery discharge event that the DC system is designed for.~~

~~If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within [12] hours (Required Action A.2).~~

BASES

ACTIONS (continued)

~~Required Action A.2 requires that the battery float current be verified as less than or equal to [2] amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial [12] hour period the battery float current is not less than or equal to [2] amps this indicates there may be additional battery problems and the battery must be declared inoperable.~~

~~----------REVIEWER'S NOTE----------~~
~~Any licensee wishing to adopt a Completion Time greater than 72 hours for Required Action A.3 will need to demonstrate that the Completion Time is appropriate for the plant in accordance with the guidance in Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications." Otherwise, the 72 hour Completion Time must be adopted.~~
~~----------~~

~~Required Action A.3 limits the restoration time for the inoperable battery charger to [72] hours. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., balance of plant non-Class 1E battery charger). The [72] hour Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.~~

3

A.1, A.2, and A.3

~~B.1, B.2.1, B.2.2, and B.2.3~~

With the required DC electrical power subsystem inoperable, the minimum required DC electrical power source is not available. Therefore, suspension of

3

~~[If two subsystems are required by LCO 3.8.10, the remaining subsystem with DC power available may be capable of supporting sufficient systems to allow continuation of [recently] irradiated fuel movement]. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend~~ movement of ~~[recently]~~ irradiated fuel assemblies, and operations involving positive reactivity additions~~)~~ that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)~~.~~ Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safety operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than ~~that~~ what

2

3

2

is required

1

5

St. Lucie – Unit 2

1

BASES

ACTIONS (continued)

would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystem~~[s]~~ and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems. (2)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystem~~s~~ should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power. (1)

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 states that Surveillances required by SR 3.8.4.1 through SR 3.8.4.3 are applicable in these MODES. See the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. FSAR, Chapter ~~[~~6~~]~~.
2. FSAR, Chapter ~~[~~15~~]~~.

(U) (2) (1)

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.5 BASES, DC SOURCES - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes have been made to be consistent with changes made to the Specifications.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

5. Typographical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.8.5, DC SOURCES - SHUTDOWN

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

ITS 3.8.6, Battery Parameters

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ELECTRICAL POWER SYSTEMS

BATTERY PARAMETERS

~~D.C. DISTRIBUTION - OPERATING~~

LIMITING CONDITION FOR OPERATION

Add proposed LCO 3.8.6

A02

LCO 3.8.6

3.8.2.3 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt D.C. bus No. 1A, 125-volt Battery bank No. 1A and a full capacity charger.

b. 125-volt D.C. bus No. 1B, 125-volt Battery bank No. 1B and a full capacity charger.

See ITS 3.8.4

See ITS 3.8.9

Applicability

APPLICABILITY: ~~MODES 1, 2, 3 and 4.~~

When associated DC electrical power subsystems are required to be OPERABLE.

A02

See ITS 3.8.9

ACTION:

a. With one of the required battery banks or busses inoperable, restore the inoperable battery bank or bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.8.4

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.3.2.a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

Add proposed ACTIONS A, B, C, D, E and F.

L01

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying indicated power availability.

See ITS 3.8.9

4.8.2.3.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

See ITS 3.8.4

a. In accordance with the Surveillance Frequency Control Program by verifying that:

SR 3.8.6.1
SR 3.8.6.2

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129-volts on float charge.

See ITS 3.8.4

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.6.3
SR 3.8.6.4
SR 3.8.6.5

b. In accordance with the Surveillance Frequency Control Program ~~and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts,~~ by verifying that: L02

SR 3.8.6.3
SR 3.8.6.5

1. The parameters in Table 4.8-2 meet the Category B limits,

2. ~~There is no visible corrosion at either terminals or connectors, and~~ L03

SR 3.8.6.4

3. ~~The average electrolyte temperature of 10% (60 cells total) of connected cells is above 50°F~~. LA01

Each battery pilot cell temperature is greater than or equal to minimum established design limits.

c. In accordance with the Surveillance Frequency Control Program by verifying that:

1. ~~The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,~~

2. ~~The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,~~

3. ~~Battery cell inter-connection resistance values are maintained at the values below:~~

~~Battery Inter-Connection Measurement Limits~~		
~~Battery Inter-Connection Type~~	~~Maximum Individual Inter-Connection Resistance~~	~~Maximum Average Inter-Connection Resistance [Battery Bank*]~~
~~Inter-Cell~~	~~≤ 150 x 10-6 ohms~~	~~≤ 50 x 10-6 ohms~~
~~Inter-Tier~~	~~≤ 200 x 10-6 ohms~~	
~~Inter-Rack~~	~~≤ 200 x 10-6 ohms~~	
~~Output Terminal~~	~~≤ 150 x 10-6 ohms~~	

L03

~~* The battery bank average interconnection resistance limit is the average of all inter-cell, inter-tier, inter-rack and output terminal connection resistance measurements for all series connections in the battery string~~

~~and,~~

4. The battery charger will supply at least 300 amperes at 140 volts for at least 6 hours.

See ITS 3.8.4

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

See ITS 3.8.4

d. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

Add proposed SR 3.8.6.6 NOTE

SR 3.8.6.6

e. In accordance with the Surveillance Frequency Control Program~~, during shutdown,~~ by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.3.2.d.

L04

L05

or a modified performance discharge test

SR 3.8.6.6

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. ~~Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.~~

See ITS 3.8.4

LA02

Add proposed SR 3.8.6.6 third Frequency

L06

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENT

	CATEGORY A(1)		CATEGORY B(2)	
Parameter	Limits for each designated pilot cell		Limits for each connected cell	Allowable(3) value for each connected cell
SR 3.8.6.3 Electrolyte Level	~~> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark~~		~~> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark~~	~~Above top of plates and not overflowing~~
SR 3.8.6.2 SR 3.8.6.5 Float Voltage	≥ ~~2.13~~ 2.07 volts	See ITS 5.5.14	≥ 2.13 volts(c)	~~≥~~ 2.07 volts
SR 3.8.6.1 ~~Specific Gravity(a)~~	~~≥ 1.195~~(b)		~~≥ 1.190 Average of all connected cells ≥ 1.200~~	~~Not more than .020 below the average of all connected cells~~ ~~Average of all connected cells ≥ 1.190(b)~~

L01 L07 L01 LA03 L08 L09 L10

greater than or equal to the minimum established design limits.

~~(a) Corrected for electrolyte temperature and level.~~

SR 3.8.6.1 (b) Or battery charging current is less than 2 amps when on charge.

(c) Corrected for average electrolyte temperature. — See ITS 5.5.14

~~(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.~~

~~(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within the limits within 7 days.~~

~~(3) With any Category B parameter not within its allowable value, declare the battery inoperable.~~

L01

ELECTRICAL POWER SYSTEMS

BATTERY PARAMETERS

~~D.C. DISTRIBUTION - SHUTDOWN~~

Add proposed LCO 3.8.6

A02

LIMITING CONDITION FOR OPERATION

LCO 3.8.6

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

1 - 125-volt D.C. bus, and

1 - 125-volt battery bank and charger supplying the above D.C. bus.

See ITS 3.8.5 and 3.8.10

Applicability

APPLICABILITY: ~~MODES 5 and 6.~~ When associated DC electrical power subsystems are required to be OPERABLE.

A02

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

See ITS 3.8.5 and 3.8.10

Add proposed ACTIONS A, B, C, D, E and F.

L01

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying indicated power availability.

See ITS 3.8.10

SR 3.8.6.1
SR 3.8.6.2
SR 3.8.6.3
SR 3.8.6.4
SR 3.8.6.5
SR 3.8.6.6

4.8.2.4.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

ELECTRICAL POWER SYSTEMS

BATTERY PARAMETERS

3/4.8.2 ~~D.C. SOURCES~~

OPERATING

LIMITING CONDITION FOR OPERATION

Add proposed LCO 3.8.6

A02

LCO 3.8.4

3.8.2.1 ~~As a minimum the following D.C. electrical sources shall be OPERABLE:~~

~~a.~~ ~~125-volt Battery bank No. 2A and a full capacity charger.~~

~~b.~~ ~~125-volt Battery bank No. 2B and a full capacity charger.~~

See ITS 3.8.4

Applicability **APPLICABILITY**: ~~MODES 1, 2, 3, and 4.~~

When associated DC electrical power subsystems are required to be OPERABLE.

A02

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 30 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.1a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

See ITS 3.8.4

Add proposed ACTIONS A, B, C, D, E and F.

L01

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

See ITS 3.8.4

a. In accordance with the Surveillance Frequency Control Program by verifying that:

SR 3.8.6.1
SR 3.8.6.2

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129-volts on float charge.

See ITS 3.8.4

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.8.6.3
SR 3.8.6.4
SR 3.8.6.5

b. In accordance with the Surveillance Frequency Control Program ~~and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts,~~ by verifying that: L02

SR 3.8.6.3
SR 3.8.6.5

1. The parameters in Table 4.8-2 meet the Category B limits,

2. ~~There is no visible corrosion at either terminals or connectors, and~~ L03

SR 3.8.6.4

3. ~~The average electrolyte temperature of 10% (60 cells total) of connected cells is above 50°F.~~ LA01

Each battery pilot cell temperature is greater than or equal to minimum established design limits.

c. In accordance with the Surveillance Frequency Control Program by verifying that:

1. ~~The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,~~

2. ~~The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,~~

3. ~~Battery cell inter-connection resistance values are maintained at the values below:~~

~~Battery Inter-Connection Measurement Limits~~		
~~Battery Inter-Connection Type~~	~~Maximum Individual Inter-Connection Resistance~~	~~Maximum Average Inter-Connection Resistance [Battery Bank*]~~
~~Inter-Cell~~	~~≤ 150 x 10-6 ohms~~	~~≤ 50 x 10-6 ohms~~
~~Inter-Tier~~	~~≤ 200 x 10-6 ohms~~	
~~Inter-Rack~~	~~≤ 200 x 10-6 ohms~~	
~~Output Terminal~~	~~≤ 150 x 10-6 ohms~~	

L03

~~* The battery bank average interconnection resistance limit is the average of all inter-cell, inter-tier, inter-rack and output terminal connection resistance measurements for all series connections in the battery string~~

~~and,~~

4. The battery charger will supply at least 300 amperes at 140 volts for at least 6 hours.

See ITS 3.8.4

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

See ITS 3.8.4

d. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

Add proposed SR 3.8.6.6 NOTE

SR 3.8.6.6

e. In accordance with the Surveillance Frequency Control Program~~, during shutdown,~~ by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1d.

L04

L05

or a modified performance discharge test

SR 3.8.6.6

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. ~~Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.~~

See ITS 3.8.4

LA02

Add proposed SR 3.8.6.6 third Frequency

L06

TABLE 4 8-2

BATTERY SURVEILLANCE REQUIREMENT

	CATEGORY A(1)		CATEGORY B(2)	
Parameter	Limits for each designated pilot cell		Limits for each connected cell	Allowable(3) value for each connected cell
SR 3.8.6.3 Electrolyte Level	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark		> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates, and not overflowing greater than or equal to the minimum established design limits.
SR 3.8.6.2 SR 3.8.6.5 Float Voltage	≥ 2.13 volts 2.07		≥ 2.13 volts(c) See ITS 5.5.14	≥ 2.07 volts ≥
			≥ 1.190	Not more than .020 below the average of all connected cells
SR 3.8.6.1 Specific Gravity(a)	≥ 1.195(b)		Average of all connected cells ≥ 1.200	Average of all connected cells ≥ 1.190(b)

L01 L07 L01 LA03 L08 L09 L10

(a) Corrected for electrolyte temperature and level.

SR 3.8.6.1 (b) Or battery charging current is less than 2 amps when on charge.

(c) Corrected for average electrolyte temperature. See ITS 5.5.14

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within the limits within 7 days.

(3) With any Category B parameter not within its allowable value, declare the battery inoperable.

L01

ELECTRICAL POWER SYSTEMS

BATTERY PARAMETERS

~~D.C. SOURCES~~

SHUTDOWN

LIMITING CONDITION FOR OPERATION

Add proposed LCO 3.8.6

A02

3.8.2.2 As a minimum, one 125-volt battery bank and a full capacity charger shall be OPERABLE.

See ITS 3.8.5

Applicability **APPLICABILITY**: ~~MODES 5 and 6.~~

When associated DC electrical power subsystems are required to be OPERABLE.

A02

ACTION:

a. With the required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, or movement of irradiated fuel; initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through a 3.58 square inch vent.

b. With the required full capacity charger inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

See ITS 3.8.5

Add proposed ACTIONS A, B, C, D, E and F.

L01

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1
SR 3.8.6.2
SR 3.8.6.3
SR 3.8.6.4
SR 3.8.6.5
SR 3.8.6.6

4.8.2.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

DISCUSSION OF CHANGES
ITS 3.8.6, BATTERY PARAMETERS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.8.2.3 and Unit 2 CTS 3.8.2.1 are applicable during MODES 1, 2, 3, and 4. Unit 1 CTS 3.8.2.4 and Unit 2 CTS 3.8.2.2 are applicable during MODES 5 and 6. ITS LCO 3.8.6 requires the battery parameters for the Train A and Train B DC electrical power subsystem batteries to be within limits when the associated DC electrical power subsystems are required to be OPERABLE. This changes the CTS by combining the requirements for the DC electrical subsystem battery parameters into one Specification and replacing the actual MODES with the phrase "When associated DC electrical power subsystems are required to be OPERABLE."

The purpose of ITS 3.8.6 is to provide battery parameter requirements for safety related batteries. This change combines Unit 1 CTS 3.8.2.3 and 3.8.2.4 and Unit 2 CTS 3.8.2.1 and 3.8.2.2 requirements for the DC electrical power subsystem battery parameters into one Specification. There are no technical changes as a result of this change since it converts the requirements into the format of the ITS. The proposed LCO states that the required battery's parameters shall be within limits. The proposed Applicability ensures the battery parameter requirements are met when the associated battery is required to be OPERABLE. Any technical changes to the battery parameters are discussed in a Discussion of Change specifically associated with that change. Any changes to the LCO and Applicability of the DC electrical power subsystem batteries are discussed in the Discussion of Changes for ITS 3.8.4 and 3.8.5. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 4.8.2.3.2.b.3 and Unit 2 4.8.2.1.b.3 require, in part, verifying the average electrolyte temperature of 10% of connected cells above

50°F. ITS SR 3.8.6.4 requires, in part, verification that each required battery pilot cell temperature is greater than or equal to minimum established design limits. The pilot cell temperature for the minimum established design limits is moved to the technical specification Bases. This changes the CTS by moving the specified temperature value for this SR to the Bases.

The removal of these details related to battery electrolyte temperature limits from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety and the removed information will be adequately controlled in the ITS Bases. The Technical Specification Bases Control Program in Chapter 5 controls changes to the Bases. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications. Furthermore, the battery and its preventive maintenance and monitoring program are under the regulatory requirements of 10 CFR 50.65. This relocation will continue to assure that the battery is maintained at current levels of performance, and that operators appropriately focus on monitoring the battery parameters for degradation.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 4.8.2.3.2.f and Unit 2 CTS 4.8.2.1.f require the performance of a battery performance test. The Surveillance requires a more frequent performance if the battery shows signs of "degradation" or has reached 85% of the service life expected for the application. The CTS further states that degradation is indicated when the battery capacity drops more than 10% from its capacity on previous performance tests or is below 90% of the manufacturer's rating. ITS SR 3.8.6.6 requires verification of the battery capacity when subjected to a performance discharge test or a modified performance discharge test. The Surveillance is also required more frequently when the battery shows degradation or has reached 85% of the expected life, but the definition of what constitutes "degradation" is not included. This changes the CTS by moving the detail on how degradation is determined from the CTS to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS SR 3.8.6.6 retains the requirement to verify the battery capacity when subjected to a performance discharge test or a modified performance discharge test. The Surveillance also requires more frequent performance when the battery shows degradation or has reached 85% of the expected life. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. The Technical Specification Bases Control Program in Chapter 5 controls changes to the Bases. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA03 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 4.8.2.3.2.b.1 and Unit 2 CTS 4.8.2.1.b require verification that battery electrolyte level is greater than the minimum level indication mark, and less than or equal to ¼ inch above maximum level indication mark for the safety related batteries. ITS SR 3.8.6.3 requires a similar Surveillance and specifies the acceptance criteria as "greater than or equal to the minimum established design limits." The minimum established design limits for battery electrolyte level will be placed in the technical specification Bases. This changes the CTS by moving the design detail for this SR from CTS to the Bases. | R1 | R1

The removal of these details related to Surveillance Requirement limits from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. This Category B value of CTS Table 4.8-2 represent appropriate monitoring levels and appropriate preventive maintenance levels for long-term battery quality and extended battery life. Removing this design detail to the Bases is acceptable because these battery parameter values will continue to be controlled at their current level and the removed information will be adequately controlled in the ITS Bases. The Technical Specification Bases Control Program in Chapter 5 controls changes to the Bases. This program provides for the evaluation of changes to ensure the Bases are properly controlled. Furthermore, the battery and its preventive maintenance and monitoring program are under the regulatory requirements of 10 CFR 50.65. This relocation will continue to assure that the battery is maintained at current levels of performance, and that operators appropriately focus on monitoring the battery parameters for degradation. This change is designated as a less restrictive removal of detail change because the Surveillance limits are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.8.2.3 and 3.8.2.4 and Unit 2 CTS 3.8.2.1 and 3.8.2.2 provide ACTIONS and associated Completion Times for when a DC electrical power subsystem battery is inoperable due to battery parameters not within limits. In addition, CTS Table 4.8.2 provide in the form of Notes (1), (2), and (3), ACTIONS and associated Completions Times when Category A or Category B parameter limits or allowable values are exceeded. In lieu of these current Actions under these conditions, ITS 3.8.6 ACTIONS provide compensatory Required Actions and associated Completion Times when battery parameters are not within limits. This changes the CTS by replacing the current ACTIONS with new compensatory ACTIONS for battery parameters not within limits.

The purpose of Unit 1 CTS 3.8.2.3 and 3.8.2.4 and Unit 2 CTS 3.8.2.1 and 3.8.2.2 ACTIONS is to provide a finite period for continued operation when a safety related battery is inoperable due to battery parameters not within limits. ITS 3.8.6 ACTIONS also allow a certain amount of time to restore battery parameters to within limits before declaring the associated battery inoperable. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions to minimize risk associated with continued operation while providing time to correct

degraded conditions. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features including the capacity and capability of remaining systems or features, time for repairs or replacement, and the low probability of a DBA occurring during the Completion Time. ACTIONS have been added to allow a short time period to restore battery parameters to within limits. The Note to the ACTIONS clarifies that separate Condition entry is allowed for each battery. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each battery parameter not within limits. Complying with the Required Actions for one inoperable battery may allow for continued operation, and a subsequent inoperable battery is governed by separate Condition entry and application of associated Required Actions. ITS 3.8.6 ACTION A covers the condition of one battery with one or more battery cells float voltage less than the specified limit and requires the performance of SR 3.8.4.1 within 2 hours, the performance of SR 3.8.6.1 within 2 hours, and restoration of the affected cell voltage to within limits within 24 hours. ITS 3.8.6 ACTION B covers the condition of one battery with float current not within the specified limit and requires the performance of SR 3.8.4.1 within 2 hours and restoration of the battery float current to within limits within 12 hours. ITS 3.8.6 ACTION C covers the condition of one battery with one or more cells electrolyte level less than minimum established design limits and requires the restoration of electrolyte level to above top of plates within 8 hours, verification that there is no evidence of leakage within 12 hours, and restoration of electrolyte level to greater than or equal to the minimum established design limits within 31 days. ITS 3.8.6 ACTION C NOTE requires Required Action C.2 to be completed if electrolyte level was below the top of plates. With electrolyte level below the top of the plates there is a potential for dryout and plate degradation; therefore, this Note requires verification of no evidence of leakage even if the level is increased above the top of the plates allowing the Condition to be exited. ITS 3.8.6 Required Action C.1 NOTE states that Required Actions C.1 and C.2 are only applicable if electrolyte level was below the top of the plates allowing 31 days to restore electrolyte level to greater than or equal to the minimum established design level. ITS 3.8.6 ACTION D covers the condition of one battery with pilot cell electrolyte temperature less than the minimum established design limits and requires the restoration of battery pilot cell temperature to greater than or equal to minimum established design limits within 12 hours. ITS 3.8.6 ACTION E covers the condition of two batteries with battery parameters not within limits and requires restoration of the battery parameters for one battery to within limits in 2 hours. ITS 3.8.6 ACTION F covers the conditions when a Required Action and associated Completion Time of any of the above ACTIONS cannot be met, if one or both batteries with one or more battery cells float voltage and float current are not within limits, or SR 3.8.6.6 is not met, and requires the immediate declaration that the associated battery is inoperable. The allowances are considered acceptable since only a short time is allowed to exist with battery parameters not within limits. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 7 – Relaxation of Surveillance Frequency)* Unit 1 CTS 4.8.2.3.2.b and Unit 2 CTS 4.8.2.1.b require the performance of several Surveillances within 7 days after a battery discharge (battery terminal voltage below 110 volts), or battery overcharge (battery terminal voltage above 150 volts). ITS 3.8.6 does not

require these Surveillances to be performed after a battery discharge or overcharge. (See DOC L03 for the discussion on deleting the battery inspection requirements from CTS. See DOC L10 for the discussion on replacing the requirement to verify the battery cell specific gravity within limits to a requirement to verify battery float current within limits.) This changes the CTS by not including a specific Surveillance Requirement to perform these tests after a discharge or overcharge.

The purpose of the Unit 1 CTS 4.8.2.3.2.b and Unit 2 CTS 4.8.2.1.b Frequency is to ensure the batteries remain OPERABLE after a severe battery discharge or overcharge. This change is acceptable because the proposed Surveillance Requirement Frequency continues to ensure an acceptable level of equipment reliability. ITS SR 3.8.6.1 requires verification that each battery float current is ≤ 2 amps at a periodic frequency in accordance with the Surveillance Frequency Control Program. The float current requirements are based on the float current indicative of a charged battery. Therefore, this Surveillance will detect a discharge condition of the battery. In addition, ITS 5.5.14, "Battery Monitoring and Maintenance Program," requires a program for battery maintenance based on the recommendations of IEEE 450. The requirement to perform these battery preventative maintenance activities are consistent with IEEE 450, and as such, will be maintained in the plant procedures implementing ITS 5.5.14. This change is designated as less restrictive because Surveillances will be performed less frequently under ITS than under CTS.

L03 *(Category 5 – Deletion of Surveillance Requirement)* Unit 1 CTS 4.8.2.3.2.b.2 and Unit 2 CTS 4.8.2.1.b.2 require verification that there is no visible corrosion at either terminals or connectors. Unit 1 CTS 4.8.2.3.2.c and Unit 2 CTS 4.8.2.1.c, in part, require verification that the cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, the cell-to-cell and terminal connections are clean, tight and coated with anti-corrosion material, and battery cell interconnection resistance values are maintained with defined limits. ITS 3.8.6 does not include these requirements for battery inspections, the removal of visible corrosion, and the verification that the battery-to-battery and terminal connections are clean, tight, and coated with anti-corrosion material. This changes the CTS by deleting the explicit battery requirements from the Technical Specifications.

The purpose of Unit 1 CTS 4.8.2.3.2.b.2 and CTS 4.8.2.3.2.c and Unit 2 CTS 4.8.2.1.b.2 and CTS 4.8.2.1.c is to ensure that proper preventative maintenance activities are performed on the safety related batteries. In accordance with ITS SR 3.0.1, when any SR is not met, the LCO is not met. This is based on the premise that SRs represent the minimum acceptable requirements for OPERABILITY of the required equipment. However, the failure to meet these specific Surveillances does not necessarily mean that the equipment is not capable of performing its safety function. When the batteries are capable of meeting ITS SR 3.8.4.1, the battery terminal voltage verification and ITS SR 3.8.4.3, the battery capacity test, they are considered to be able to meet their safety function. The Surveillances that are proposed to be deleted are considered preventative maintenance activities and are not considered the minimum acceptable requirements for OPERABILITY of the batteries. This change is acceptable because the SR requirements proposed in ITS 3.8.4

continue to ensure that the batteries are maintained consistent with the safety analyses and licensing basis. In addition, ITS 5.5.14, "Battery Monitoring and Maintenance Program," requires a program for battery maintenance based on the recommendations of IEEE 450. The requirement to perform these battery preventative maintenance activities are consistent with IEEE 450, and as such, will be maintained in plant procedures implementing ITS 5.5.14. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L04 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS surveillance requirement 4.8.2.3.2.e and Unit 2 CTS surveillance requirement 4.8.2.1.e require the performance of discharge tests verifying battery capacity in accordance with the Surveillance Frequency Control Program during shutdown. ITS SR 3.8.6.6, in part, requires the verification of battery capacity when subjected to a performance discharge test. A Note modifies the ITS SR. The Note states “This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR". This changes the CTS by allowing portions of the surveillance to be performed in MODES 1, 2, 3, or 4 (to confirm OPERABILITY) when an assessment determines safe plant operation can be maintained. | R1

The purpose of the CTS requirement to perform the required surveillance during shutdown conditions is to assure plant safety. The ITS SR note allows the SR to be conducted during power operation but only if the performance of the SR does not reduce plant safety. The proposed change is acceptable because it continues to adequately verify that the equipment used to meet the LCO can perform its required functions while assuring the plant is operated safely. The allowance to perform the SR in MODE 1, 2, 3, or 4 is accompanied by the requirement to assess the impact on safety prior to performing the SR. The requirement to assess the impact on the safe operation of the plant provides additional assurance that plant safety will not be decreased. Thus, the proposed ITS relaxation for performing this SR at power is balanced by the additional requirement to assure safe plant operation. The proposed ITS requirement is consistent with the ISTS wording for this requirement. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L05 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.8.2.3.2.e and Unit 2 CTS 4.8.2.1.e require the performance of a battery performance discharge test on the batteries. ITS SR 3.8.6.6 requires the performance of a performance discharge test or a modified performance discharge test. This changes the CTS by adding the allowance to perform a modified performance discharge test instead of the performance discharge test.

The purpose of Unit 1 CTS 4.8.2.3.2.e and Unit 2 CTS 4.8.2.1.e is to determine overall battery degradation due to age and usage. A performance test, as defined in IEEE 450, is a constant-current or constant-power capacity test made on a battery after it has been in service, to detect any change in the capacity. A modified performance test, as defined in IEEE 450, is a test, in the "as found"

condition, of battery capacity and the ability of the battery to satisfy the duty cycle. Both tests, performance discharge test or modified performance discharge test, monitor the battery capacity. IEEE 450, as endorsed by RG 1.129-2007, states that a modified performance test is a test of battery capacity using a constant current, modified by increasing the current to bound the currents in the duty cycle. Deviations from the constant-current test, which increase the current, are acceptable. IEEE 450 also states that a modified performance test can be used in lieu of a service test and/or a performance test at any time. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are acceptable for verification that the equipment used to meet the LCO can perform its required functions. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L06 *(Category 7 – Relaxation of Surveillance Frequency)* Unit 1 CTS 4.8.2.3.2.f and Unit 2 CTS 4.8.2.1.f require an increased Frequency from that in Unit 1 CTS 4.8.2.3.2.e and Unit 1 CTS 4.8.2.1.e for battery performance tests if the battery shows signs of degradation or has reached 85% of its expected service life. ITS SR 3.8.6.6 provides two Frequencies when a battery has reached 85% of its expected service life based on whether the battery's remaining capacity is less than 100% or not. If the battery's remaining capacity is less than 100% of the manufactures rating, the required discharge test is performed every 12 months (annually). If the battery's remaining capacity is greater than or equal to 100% of the manufacturer's rating, then the required discharge test is performed every 24 months. This changes the CTS by relaxing the Frequency of required discharge tests when the battery has reached 85% of its expected service life.

The purpose of Unit 1 CTS 4.8.2.3.2.f and Unit 2 CTS 4.8.2.1.f is to verify the battery capacity is acceptable on batteries that show signs of degradation or have reached 85% of the service life. ITS 3.8.6.6 provides an alternative Frequency if the battery that has reached 85% of its service life but has not shown signs of degradation and has a capacity of greater than or equal to 100% of the manufacture's rating. This relaxed frequency is acceptable because the battery has not shown signs of degradation, retains greater than or equal to the manufacturer's rated capacity, and is being tested at a more frequent periodicity than a battery that has not reached 85% of the service life expected. Also, IEEE 450 states that if the battery has reached 85% of service life, delivers a capacity of 100% or greater of the manufacturer's rated capacity, and has shown no signs of degradation, performance testing at two-year intervals is acceptable until the battery shows signs of degradation. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L07 *(Category 5 – Deletion of Surveillance Requirement)* Unit 1 CTS 4.8.2.3.2.a.1 and Unit 2 CTS 4.8.2.1.a.1 require verifying the parameters in Table 4.8-1a meet Category A limits in accordance with the Surveillance Frequency Control Program. CTS Table 4.8-2 Category A contains the parameter 'Electrolyte Level' with the limit for each designated pilot cell set at "> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark." Unit 1 CTS 4.8.2.3.2.b.1 and Unit 2 CTS 4.8.2.1.a.1 require verifying the parameters in Table 4.8-2 meet Category B limits in accordance with the Surveillance Frequency Control

Program. CTS Table 4.8-2 Category B contains the parameter 'Electrolyte Level' with the limit for each connected cell set at "> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark." In addition, the Category B electrolyte level Allowable Value for each connected cell (which includes the pilot cells) is above the top of plates, and not overflowing. ITS SR 3.8.6.3 requires verifying each battery connected cell electrolyte level is greater than or equal to minimum established design limits in accordance with the Surveillance Frequency Control Program. This changes the CTS by deleting the requirement to verify the pilot cells electrolyte level is within limits.

The purpose of CTS Category A limit on pilot cell electrolyte level and its related Frequency is to represent appropriate monitoring levels and appropriate preventive maintenance levels for long-term battery quality and extended battery life. The definition of Limiting Condition for Operation (LCO) presented in 10 CFR 50.36 states that LCOs are "the lowest functional capability or performance levels of equipment required for safe operation of the facility." As such, the Category A value for cell electrolyte level does not reflect the 10 CFR 50.36 criteria for LCOs. ITS 5.5.14, Battery Monitoring and Maintenance Program," requires a program providing controls for battery restoration and maintenance that shall be in accordance with IEEE 450 as endorsed by Regulatory Guide 1.129, with exceptions as applicable. IEEE 450 contains, in part, guidance on monitoring electrolyte level with the intention of providing recommended maintenance, test schedules, and testing procedures that can be used to optimize the life and performance of permanently installed, vented lead-acid storage batteries used for standby power applications. These values and actions associated with restoration are being replaced by a licensee-controlled program, required and described in TS Section 5.5, "Programs and Manuals," and titled the "Battery Monitoring and Maintenance Program." This provides adequate assurance that necessary battery parameter values will continue to be controlled and actions will be implemented if the battery parameter values are not met. Furthermore, the battery and its preventive maintenance and monitoring program are under the regulatory requirements of 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L08 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.8.2.3.2.a.1 and Unit 2 CTS 4.8.2.1.a.1 require verification that the pilot cell voltage is ≥ 2.13 V. ITS SR 3.8.6.2 requires the verification that each pilot cell voltage is ≥ 2.07 V. ITS 3.8.6 ACTION A addresses the condition in which one or more batteries with one or more battery cells float voltage less than 2.07 V. Once ACTION A has been entered, the battery cell is considered degraded and the Required Actions are to perform SR 3.8.4.1 and SR 3.8.6.1 within 2 hours. This changes the CTS by reducing the acceptance criterion for the pilot cell voltage limit from ≥ 2.13 V to ≥ 2.07 V.

The purpose of the proposed Surveillance limit in ITS SR 3.8.6.2 is to ensure the cell voltages are greater than or equal to the short-term absolute minimum voltage. Optimal long-term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer. This provides adequate over-potential,

which limits the formation of lead sulfate and self discharge which could eventually render the battery inoperable. Float voltages in this range or less, but greater than 2.07 V per cell, are addressed in the Battery Monitoring and Maintenance Program. Furthermore, the Battery Monitoring and Maintenance Program includes actions to restore battery cells with float voltage less than 2.13 V and actions to verify that the remaining cells are greater than or equal to 2.07 V when a cell or cells have been found to be less than 2.13 V. A cell voltage of 2.07 V or below under float conditions may indicate internal cell problems while prolonged operation of cells below 2.13 V has the potential to reduce the life expectancy of cells. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. At the lower proposed voltage, the cell can still perform its function. The battery is considered OPERABLE when the battery voltage on float is greater than or equal to the minimum establish voltage of ITS SR 3.8.4.1. This change is acceptable since ITS 5.5.14, "Battery Monitoring and Maintenance Program," requires actions to be taken to restore battery cells with float voltage < 2.13 V. This program helps ensure the cell voltage will not approach the limit of 2.07 V and that the minimum established voltage of ITS SR 3.8.4.1 is maintained. This change is designated as less restrictive because a less stringent Surveillance Requirement acceptance criterion is being applied in the ITS than applied in the CTS.

L09 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* Unit 1 CTS 4.8.2.3.2.b.1 and Unit 2 CTS 4.8.2.1.b.1 require verification that each connected cell voltage is > 2.07 V in accordance with the Surveillance Frequency Control Program. ITS SR 3.8.6.5 requires the verification that each connected cell voltage is ≥ 2.07 V. This changes the CTS by reducing the acceptance criteria for each connected cell voltage limit from > 2.07 V to ≥ 2.07 V.

The purpose of Unit 1 CTS 4.8.2.3.2.b.1 and Unit 2 CTS 4.8.2.1.b.1 is to verify each connected cell voltage > 2.07 V, thereby establishing a minimum value for battery OPERABILITY. The purpose of the ITS 3.8.6.5 requirement to verify each connected cell voltage is ≥ 2.07 V is also to establish a minimum value for battery OPERABILITY. Optimal long-term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to 2.06 V per cell. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than 2.07 V per cell, are addressed in the Battery Monitoring and Maintenance Program. Furthermore, the Battery Monitoring and Maintenance Program includes actions to restore battery cells with float voltage less than 2.13 V and actions to verify that the remaining cells are greater than or equal to 2.07 V when a cell or cells have been found to be less than 2.13 V. The 2.07 V per individual cell limit reflects the OPERABILITY limit for the batteries. With all battery cells at or above 2.07 V, there is adequate assurance that that the terminal voltage is at an acceptable threshold for establishing battery OPERABILITY. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L10 *(Category 5 – Deletion of Surveillance Requirement)* Unit 1 CTS 4.8.2.3.2.a.1 and Unit 2 CTS 4.8.2.1.a.1 require verification the pilot cell specific gravity is within the Category A limits of Table 4.8-2, as modified by footnote (a)). Unit 1 CTS 4.8.2.3.2.b.1 and Unit 2 CTS 4.8.2.1.b.1 require verification that each connected cell specific gravity is within the Category B limits of Table 4.8-2. As indicated in CTS Table 4.8-2 (footnote (a)), the specific gravity limit must be corrected for electrolyte temperature and level. ITS 3.8.6 does not include these Surveillances. This changes the CTS by deleting the Surveillances to verify battery cell specific gravity.

The purpose of Unit 1 CTS 4.8.2.3.2.a.1 and CTS 4.8.2.3.2.b.1 and Unit 2 CTS 4.8.2.1.a.1 and CTS 4.8.2.1.b.1 is to ensure the state of charge of each battery cell. This change is acceptable because the deleted Surveillance Requirements are not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. While the specified Surveillances have been deleted, the alternative Surveillance of CTS Table 4.8-2 footnote (b), to verify battery charging current < 2 amps is retained in ITS as SR 3.8.6.1, with modifications to the acceptance criteria to reflect the information provided by the battery manufacturer. IEEE 450 states that the most accurate indicator of return to full charge is a stabilized charging or float current. Specific gravity readings may not be accurate when the battery is on charge following a discharge. Therefore, ITS SR 3.8.6.1 gives a better indication of the overall battery condition. This change is designated as less restrictive because Surveillances which are required in CTS will not be required in ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Parameters

3.8.2.3
3.8.2.4

LCO 3.8.6 Battery parameters for the Train A and Train B electrical power subsystem batteries shall be within limits.

Applicability

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 A. One ~~[or two]~~ batter[y][~~ies on one subsystem]~~ with one or more battery cells float voltage < [~~2.07~~] V.	A.1 Perform SR 3.8.4.1. AND A.2 Perform SR 3.8.6.1. AND A.3 Restore affected cell voltage ≥ [~~2.07~~] V.	2 hours 2 hours 24 hours
DOC L01 B. One ~~[or two]~~ batter[y][~~ies on one subsystem]~~ with float current > [~~2~~] amps.	B.1 Perform SR 3.8.4.1. AND B.2 Restore battery float current to ≤ [~~2~~] amps.	2 hours [~~12~~] hours

(2) (2) (2) (2)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 C. ----------NOTE---------- Required Action C.2 shall be completed if electrolyte level was below the top of plates. ---------- One ~~[or two]~~ batter~~[~~y~~][ies on one subsystem]~~ with one or more cells electrolyte level less than minimum established design limits.	----------NOTE---------- Required Actions C.1 and C.2 are only applicable if electrolyte level was below the top of plates. ---------- C.1 Restore electrolyte level to above top of plates. AND C.2 Verify no evidence of leakage. AND C.3 Restore electrolyte level to greater than or equal to minimum established design limits.	8 hours (2) 12 hours 31 days
DOC L01 D. One ~~[or two]~~ batter~~[~~y~~][ies on one subsystem]~~ with pilot cell electrolyte temperature less than minimum established design limits.	D.1 Restore battery pilot cell temperature to greater than or equal to minimum established design limits.	12 hours (2)
DOC L01 E. ~~One or more~~ batteries ~~in redundant subsystems~~ with battery parameters not within limits.	E.1 Restore battery parameters for batter~~ies in~~ one ~~subsystem~~ to within limits.	2 hours (1)

Two

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CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR One [or ~~two~~] batter[y][ies ~~on one subsystem~~] with one or more battery cells float voltage < [2.07] V and float current > [2] amps.	F.1 Declare associated battery inoperable.	Immediately

both

OR

SR 3.8.6.6 not met.

2

3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.2.3.2.a.1 4.8.2.4.2 SR 3.8.6.1	----------NOTE---------- Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. ---------- Verify each battery float current is ≤ [2] amps.	[~~7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

2

1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.8.2.3.2.a.1 4.8.2.4.2	SR 3.8.6.2	Verify each battery pilot cell float voltage is ≥ ~~[~~2.07~~]~~ V.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)
4.8.2.3.2.b.1 4.8.2.4.2	SR 3.8.6.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)
4.8.2.3.2.b.3 4.8.2.4.2	SR 3.8.6.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	~~[~~ 31 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)
4.8.2.3.2.b.1 4.8.2.4.2	SR 3.8.6.5	Verify each battery connected cell float voltage is ≥ ~~[~~2.07~~]~~ V.	~~[~~ 92 days ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.6	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, portions of this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ----------	
DOC L04 4.8.2.3.2.e 4.8.2.3.2.f 4.8.2.4.2 DOC L05	Verify battery capacity is ≥ [80%] of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	[60 months OR In accordance with the Surveillance Frequency Control Program] (2) AND 12 months when battery shows degradation, or has reached [85]% of the expected life with capacity < 100% of manufacturer's rating (2) AND DOC L06 24 months when battery has reached [85]% of the expected life with capacity ≥ 100% of manufacturer's rating (2)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Parameters

3.8.2.3
3.8.2.4

LCO 3.8.6 Battery parameters for the Train A and Train B electrical power subsystem batteries shall be within limits.

Applicability

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 A. One ~~[or two]~~ batter[y]~~[ies on one subsystem]~~ with one or more battery cells float voltage < ~~[~~2.07~~]~~ V.	A.1 Perform SR 3.8.4.1. AND A.2 Perform SR 3.8.6.1. AND A.3 Restore affected cell voltage ≥ ~~[~~2.07~~]~~ V.	2 hours 2 hours 24 hours
DOC L01 B. One ~~[or two]~~ batter[y]~~[ies on one subsystem]~~ with float current > ~~[~~2~~]~~ amps.	B.1 Perform SR 3.8.4.1. AND B.2 Restore battery float current to ≤ ~~[~~2~~]~~ amps.	2 hours ~~[~~12~~]~~ hours

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 C. ----------NOTE---------- Required Action C.2 shall be completed if electrolyte level was below the top of plates. ---------- One ~~[or two]~~ batter~~[~~y~~][ies on one subsystem]~~ with one or more cells electrolyte level less than minimum established design limits.	----------NOTE---------- Required Actions C.1 and C.2 are only applicable if electrolyte level was below the top of plates. ---------- C.1 Restore electrolyte level to above top of plates. AND C.2 Verify no evidence of leakage. AND C.3 Restore electrolyte level to greater than or equal to minimum established design limits.	8 hours (2) 12 hours 31 days
DOC L01 D. One ~~[or two]~~ batter~~[~~y~~][ies on one subsystem]~~ with pilot cell electrolyte temperature less than minimum established design limits.	D.1 Restore battery pilot cell temperature to greater than or equal to minimum established design limits.	12 hours (2)
DOC L01 E. ~~One or more~~ Two batteries ~~in redundant subsystems~~ with battery parameters not within limits.	E.1 Restore battery parameters for batter~~ies in~~ one ~~subsystem~~ to within limits. (y)	2 hours (1)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. OR One [or ~~two~~] batter[y][ies ~~on one subsystem~~] with one or more battery cells float voltage < [2.07] V and float current > [2] amps.	F.1 Declare associated battery inoperable.	Immediately

both

OR

SR 3.8.6.6 not met.

2

3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.2.3.2.a.1 4.8.2.4.2 SR 3.8.6.1	----------NOTE---------- Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. ----------	
	Verify each battery float current is ≤ [2] amps.	[~~7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

2

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
4.8.2.3.2.a.1 4.8.2.4.2	SR 3.8.6.2	Verify each battery pilot cell float voltage is ≥ [2.07] V.	[31 days OR In accordance with the Surveillance Frequency Control Program] 2
4.8.2.3.2.b.1 4.8.2.4.2	SR 3.8.6.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	[31 days OR In accordance with the Surveillance Frequency Control Program] 2
4.8.2.3.2.b.3 4.8.2.4.2	SR 3.8.6.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	[31 days OR In accordance with the Surveillance Frequency Control Program] 2
4.8.2.3.2.b.1 4.8.2.4.2	SR 3.8.6.5	Verify each battery connected cell float voltage is ≥ [2.07] V.	[92 days OR In accordance with the Surveillance Frequency Control Program] 2

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.6	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, portions of this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ----------	
DOC L04 4.8.2.3.2.e 4.8.2.3.2.f 4.8.2.4.2 DOC L05	Verify battery capacity is ≥ [80%] of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	[60 months OR In accordance with the Surveillance Frequency Control Program] (2) AND 12 months when battery shows degradation, or has reached [85]% of the expected life with capacity < 100% of manufacturer's rating (2) AND DOC L06 24 months when battery has reached [85]% of the expected life with capacity ≥ 100% of manufacturer's rating (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.6, BATTERY PARAMETERS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. ISTS SR 3.8.6.6 requires a battery performance discharge or modified performance discharge test to be performed and provides acceptance criteria. However, no ACTION is provided in the ISTS 3.8.6 ACTIONS for when this SR is not met. Thus in the ISTS, LCO 3.0.3 would have to be entered. To preclude an LCO 3.0.3 entry, ISTS 3.8.6 Condition F has been modified to cover the case when SR 3.8.6.6 is not met. ACTION F will require the associated battery to be declared inoperable. This is also consistent with the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Parameters

BASES

BACKGROUND

This LCO delineates the limits on battery float current as well as electrolyte temperature, level, and float voltage for the DC power subsystem batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3. 8.4, "DC Sources - Operating," and LCO 3.8.5, "DC Sources - Shutdown." In addition to the limitations of this Specification, the ~~[licensee controlled program]~~ also implements a program specified in Specification 5.5.1~~6~~ for monitoring various battery parameters.

Battery Monitoring and Maintenance Program

The battery cells are of flooded lead acid construction with a nominal specific gravity of ~~[~~1.215~~]~~. This specific gravity corresponds to an open circuit battery voltage of approximately ~~120~~ V for ~~[58]~~ cell battery (i.e., cell voltage of ~~[2.065]~~ volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. ~~Once fully charged with its open circuit voltage ≥ [2.065] Vpc, the battery cell will maintain its capacity for [30] days without further charging per manufacturer's instructions.~~ Optimal long term performance however, is obtained by maintaining a float voltage ~~[~~2.20 to 2.25~~]~~ Vpc. This provides adequate over-potential which limits the formation of lead sulfate and self discharge. The nominal float voltage of ~~[2.22]~~ Vpc corresponds to a total float voltage output of ~~[128.8]~~ V for a ~~[58]~~ cell battery ~~as discussed in the FSAR, Chapter [8] (Ref. 2)~~.

123.8
60
2.063
recommended
2.20 to 2.25
132 V to 135

APPLICABLE SAFETY ANALYSES

U

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 3) and Chapter ~~[~~15~~]~~ (Ref. 4), assume Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least one subsystem of DC sources OPERABLE during accident conditions, in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power and

b. A worst-case single failure.

Battery parameters satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Battery parameter limits are conservatively established, allowing continued DC electrical system function even with limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the ~~[licensee controlled program]~~ is conducted as specified in Specification 5.5.1~~6~~.

Battery Monitoring and Maintenance Program

4

2

3

APPLICABILITY

The battery parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery parameter limits are only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

ACTIONS

Insert 1

1

A.1, A.2, and A.3

battery

With one or more cells in one ~~or more batteries in one subsystem~~ < ~~[~~2.07~~]~~ V, the battery cell is degraded. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage (SR 3.8.4.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.6.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells in one or more batteries < ~~[~~2.07~~]~~ V, and continued operation is permitted for a limited period up to 24 hours.

6

2

2

Since the Required Actions only specify "perform," a failure of SR 3.8.4.1 or SR 3.8.6.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If SR 3.8.6.1 is failed then there is no assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

B.1 and B.2

battery

One ~~or more batteries in one subsystem~~ with float current > ~~[~~2~~]~~ amps indicates that a partial discharge of the battery capacity has occurred. This may be due to a temporary loss of a battery charger or possibly due to one or more battery cells in a low voltage condition reflecting some loss of capacity. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage. If the terminal voltage is found to be less than the minimum established float voltage there are two possibilities, the battery charger is inoperable or is

6

2

1

Insert 1

The Note to the ACTIONS clarifies that separate Condition entry is allowed for each battery. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each battery parameter not within limits. Complying with the Required Actions for one inoperable battery may allow for continued operation, and a subsequent inoperable battery is governed by separate Condition entry and application of associated Required Actions.

BASES

ACTIONS (continued)

operating in the current limit mode. Condition A addresses charger inoperability. If the charger is operating in the current limit mode after 2 hours that is an indication that the battery has been substantially discharged and likely cannot perform its required design functions. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within ~~[~~12~~]~~ hours (Required Action B.2). The battery must therefore be declared inoperable. 2

If the float voltage is found to be satisfactory but there are one or more battery cells with float voltage less than ~~[~~2.07~~]~~ V, the associated "OR" statement in Condition F is applicable and the battery must be declared inoperable immediately. If float voltage is satisfactory and there are no cells less than ~~[~~2.07~~]~~ V there is good assurance that, within ~~[~~12~~]~~ hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to a temporary loss of the battery charger. 2 2

~~REVIEWER'S NOTE~~

~~A plant that cannot meet the 12 hour Completion Time due to an inherent battery charging characteristic can propose an alternate time equal to 2 hours plus the time experienced to accomplish the exponential charging current portion of the battery charge profile following the service test (SR 3.8.4.3).~~ 5

A discharged battery with float voltage (the charger setpoint) across its terminals indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within ~~[~~12~~]~~ hours, avoiding a premature shutdown with its own attendant risk. 2

If the condition is due to one or more cells in a low voltage condition but still greater than ~~[~~2.07~~]~~ V and float voltage is found to be satisfactory, this is not indication of a substantially discharged battery and ~~[~~12~~]~~ hours is a reasonable time prior to declaring the battery inoperable. 2

BASES

ACTIONS (continued)

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.4.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.4.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

C.1, C.2, and C.3

battery

With one ~~or more batteries in one subsystem~~ with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established. 6

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. Required Actions C.1 and C.2 address this potential (as well as provisions in Specification 5.5.1~~6~~, Battery Monitoring and Maintenance Program). They are modified by a Note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates. The Required Action C.2 requirement to verify that there is no leakage by visual inspection and the Specification 5.5.1~~6~~.b item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from IEEE Standard 450. They are performed following the restoration of the electrolyte level to above the top of the plates. Based on the results of the manufacturer's recommended testing, the batter[y][~~ies~~] may have to be declared inoperable and the affected cell[~~s~~] replaced. 4 4 3 , 1 2

D.1

battery

With one ~~or more batteries in one subsystem~~ with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met. 6

BASES

ACTIONS (continued)

E.1

two

With ~~one or more~~ batteries ~~in redundant subsystems~~ with battery parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the batteries can still perform their required function, given that redundant batteries are involved. With redundant batteries involved this potential could result in a total loss of function on multiple systems that rely upon the batteries. The longer Completion Times specified for battery parameters on ~~non-redundant batteries~~ not within limits are therefore not appropriate, and the parameters must be restored to within limits on at least one ~~subsystem~~ within 2 hours.

6

one battery

1

battery

F.1

With one or more batteries with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, D, or E, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one or ~~more~~ batteries ~~in one subsystem~~ with one or more battery cells float voltage less than ~~[~~2.07~~]~~ V and float current greater than ~~[~~2~~]~~ amps indicates that the battery capacity may not be sufficient to perform the intended functions. The battery must therefore be declared inoperable immediately.

or failure of the battery performance discharge test (SR 3.8.6.6),

1

both

2

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450 (Ref. 1). ~~[The 7-day Frequency is consistent with IEEE-450 (Ref. 1).~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 5 2

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. When this float voltage is not maintained the Required Actions of LCO 3.8.4 ACTION A are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of ~~[~~2~~]~~ amps is established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained. 2

SR 3.8.6.2 and SR 3.8.6.5

allowable

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum ~~established design~~ limit~~s~~ provided by the battery manufacturer, which corresponds to ~~[130.5] V at the battery terminals~~, or ~~[2.25]~~ Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than ~~[~~2.07~~]~~ Vpc, are addressed in Specification 5.5.1~~6~~. SRs 3.8.6.2 and 3.8.6.5 require verification that the cell float voltages are equal to or greater than the short term absolute minimum voltage of ~~[~~2.07~~]~~ V. ~~[The Frequency for cell voltage verification every 31 days for pilot cell and 92 days for each connected cell is consistent with IEEE-450 (Ref. 1).~~

127.2 V for a 60 cell battery

2.12

4

1 2 3

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 2

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 5 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

Electrolyte level is monitored as part of the "Battery Monitoring and Maintenance Program" (ITS 5.5.14). Level is maintained above the top of the plates with Action taken when level is ≤ ¼" above the maximum level indication mark.

SR 3.8.6.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability. ~~[The Frequency of 31days is consistent with IEEE-450 (Ref. 1).~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

5

~~]~~ 2

SR 3.8.6.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., ~~[40]~~ 50°F). 2 Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity. ~~[The Frequency of 31 days is consistent with IEEE-450 (Ref. 1).~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

5

~~]~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.6

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as-found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.6.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.4.3.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

It may consist of just two rates; for instance the one minute rate for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test must remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 1) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this ~~[~~80~~]~~% limit. 2

~~[The Surveillance Frequency for this test is normally 60 months.~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 1), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ ~~[~~10%~~]~~ below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 1).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1~~ or 2~~ is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1~~ or 2~~. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

BASES

REFERENCES

1. IEEE-450.

2. FSAR, Chapter ~~[~~8~~]~~.

3. FSAR, Chapter ~~[~~6~~]~~.

4. FSAR, Chapter ~~[~~15~~]~~.

5. IEEE-485-~~[~~1983~~], June 1983~~.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Parameters

BASES

BACKGROUND

This LCO delineates the limits on battery float current as well as electrolyte temperature, level, and float voltage for the DC power subsystem batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3. 8.4, "DC Sources - Operating," and LCO 3.8.5, "DC Sources - Shutdown." In addition to the limitations of this Specification, the ~~[licensee controlled program]~~ also implements a program specified in Specification 5.5.1~~6~~ for monitoring various battery parameters.

Battery Monitoring and Maintenance Program

The battery cells are of flooded lead acid construction with a nominal specific gravity of ~~[~~1.215~~]~~. This specific gravity corresponds to an open circuit battery voltage of approximately ~~120~~ V for ~~[58]~~ cell battery (i.e., cell voltage of ~~[2.065]~~ volts per cell (Vpc)). The open circuit voltage is the voltage maintained when there is no charging or discharging. ~~Once fully charged with its open circuit voltage ≥ [2.065] Vpc, the battery cell will maintain its capacity for [30] days without further charging per manufacturer's instructions.~~ Optimal long term performance however, is obtained by maintaining a float voltage ~~[~~2.20 to 2.25~~]~~ Vpc. This provides adequate over-potential which limits the formation of lead sulfate and self discharge. The nominal float voltage of ~~[2.22]~~ Vpc corresponds to a total float voltage output of ~~[128.8]~~ V for a ~~[58]~~ cell battery ~~as discussed in the FSAR, Chapter [8] (Ref. 2)~~.

123.8
60
2.063
recommended
2.20 to 2.25
132 V to 135

APPLICABLE SAFETY ANALYSES

U

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[6]~~ (Ref. 3) and Chapter ~~[15]~~ (Ref. 4), assume Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least one subsystem of DC sources OPERABLE during accident conditions, in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power and

b. A worst-case single failure.

Battery parameters satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Battery parameter limits are conservatively established, allowing continued DC electrical system function even with limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the ~~[licensee controlled program]~~ is conducted as specified in Specification 5.5.1~~6~~.

Battery Monitoring and Maintenance Program

4

2

3

APPLICABILITY

The battery parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery parameter limits are only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

ACTIONS

Insert 1

1

A.1, A.2, and A.3

battery

With one or more cells in one ~~or more batteries in one subsystem~~ < ~~[~~2.07~~]~~ V, the battery cell is degraded. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage (SR 3.8.4.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.6.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells in one or more batteries < ~~[~~2.07~~]~~ V, and continued operation is permitted for a limited period up to 24 hours.

6

2

2

Since the Required Actions only specify "perform," a failure of SR 3.8.4.1 or SR 3.8.6.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If SR 3.8.6.1 is failed then there is no assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

B.1 and B.2

battery

One ~~or more batteries in one subsystem~~ with float current > ~~[~~2~~]~~ amps indicates that a partial discharge of the battery capacity has occurred. This may be due to a temporary loss of a battery charger or possibly due to one or more battery cells in a low voltage condition reflecting some loss of capacity. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage. If the terminal voltage is found to be less than the minimum established float voltage there are two possibilities, the battery charger is inoperable or is

6

2

1

Insert 1

The Note to the ACTIONS clarifies that separate Condition entry is allowed for each battery. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each battery parameter not within limits. Complying with the Required Actions for one inoperable battery may allow for continued operation, and a subsequent inoperable battery is governed by separate Condition entry and application of associated Required Actions.

BASES

ACTIONS (continued)

operating in the current limit mode. Condition A addresses charger inoperability. If the charger is operating in the current limit mode after 2 hours that is an indication that the battery has been substantially discharged and likely cannot perform its required design functions. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within ~~[~~12~~]~~ hours (Required Action B.2). The battery must therefore be declared inoperable. 2

If the float voltage is found to be satisfactory but there are one or more battery cells with float voltage less than ~~[~~2.07~~]~~ V, the associated "OR" statement in Condition F is applicable and the battery must be declared inoperable immediately. If float voltage is satisfactory and there are no cells less than ~~[~~2.07~~]~~ V there is good assurance that, within ~~[~~12~~]~~ hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to a temporary loss of the battery charger. 2 2

~~REVIEWER'S NOTE~~

~~A plant that cannot meet the 12 hour Completion Time due to an inherent battery charging characteristic can propose an alternate time equal to 2 hours plus the time experienced to accomplish the exponential charging current portion of the battery charge profile following the service test (SR 3.8.4.3).~~ 5

A discharged battery with float voltage (the charger setpoint) across its terminals indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within ~~[~~12~~]~~ hours, avoiding a premature shutdown with its own attendant risk. 2

If the condition is due to one or more cells in a low voltage condition but still greater than ~~[~~2.07~~]~~ V and float voltage is found to be satisfactory, this is not indication of a substantially discharged battery and ~~[~~12~~]~~ hours is a reasonable time prior to declaring the battery inoperable. 2

BASES

ACTIONS (continued)

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.4.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.4.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

C.1, C.2, and C.3

battery

With one ~~or more batteries in one subsystem~~ with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established. 6

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. Required Actions C.1 and C.2 address this potential (as well as provisions in Specification 5.5.1~~6~~, Battery Monitoring and Maintenance Program). They are modified by a Note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates. The Required Action C.2 requirement to verify that there is no leakage by visual inspection and the Specification 5.5.1~~6~~.b item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from IEEE Standard 450. They are performed following the restoration of the electrolyte level to above the top of the plates. Based on the results of the manufacturer's recommended testing, the batter[y]~~[ies]~~ may have to be declared inoperable and the affected cell~~[s]~~ replaced. 4 4 3 , 1 2

D.1

battery

With one ~~or more batteries in one subsystem~~ with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met. 6

BASES

ACTIONS (continued)

E.1

two

With ~~one or more~~ batteries ~~in redundant subsystems~~ with battery parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the batteries can still perform their required function, given that redundant batteries are involved. With redundant batteries involved this potential could result in a total loss of function on multiple systems that rely upon the batteries. The longer Completion Times specified for battery parameters on ~~non-redundant batteries~~ not within limits are therefore not appropriate, and the parameters must be restored to within limits on at least one ~~subsystem~~ within 2 hours.

6

one battery

1

battery

F.1

With one or more batteries with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, D, or E, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one or ~~more~~ batteries ~~in one subsystem~~ with one or more battery cells float voltage less than ~~[~~2.07~~]~~ V and float current greater than ~~[~~2~~]~~ amps indicates that the battery capacity may not be sufficient to perform the intended functions. The battery must therefore be declared inoperable immediately.

or failure of the battery performance discharge test (SR 3.8.6.6),

1

both

2

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450 (Ref. 1). ~~[The 7-day Frequency is consistent with IEEE-450 (Ref. 1).~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. When this float voltage is not maintained the Required Actions of LCO 3.8.4 ACTION A are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of ~~[~~2~~]~~ amps is established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained.

SR 3.8.6.2 and SR 3.8.6.5

allowable

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum ~~established design~~ limit~~s~~ provided by the battery manufacturer, which corresponds to ~~[130.5] V at the battery terminals~~, or ~~[2.25]~~ Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than ~~[~~2.07~~]~~ Vpc, are addressed in Specification 5.5.1~~6~~. SRs 3.8.6.2 and 3.8.6.5 require verification that the cell float voltages are equal to or greater than the short term absolute minimum voltage of ~~[~~2.07~~]~~ V. ~~[The Frequency for cell voltage verification every 31 days for pilot cell and 92 days for each connected cell is consistent with IEEE-450 (Ref. 1).~~

127.2 V for a 60 cell battery

2.12

4

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

Electrolyte level is monitored as part of the "Battery Monitoring and Maintenance Program" (ITS 5.5.14). Level is maintained above the top of the plates with Action taken when level is ≤ ¼" above the maximum level indication mark.

SR 3.8.6.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability. ~~[The Frequency of 31days is consistent with IEEE-450 (Ref. 1).~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

5

~~]~~ 2

SR 3.8.6.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., ~~[40]~~ 50°F). 2 Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity. ~~[The Frequency of 31 days is consistent with IEEE-450 (Ref. 1).~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

5

~~]~~ 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.6

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as-found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.6.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.4.3.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

It may consist of just two rates; for instance the one minute rate for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test must remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 1) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this ~~[~~80~~]~~% limit. (2)

~~[The Surveillance Frequency for this test is normally 60 months.~~ (2)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~[REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 1), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ ~~[~~10%~~]~~ below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 1).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 ~~or 2~~ is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 ~~or 2~~. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

BASES

REFERENCES

1. IEEE-450.
2. FSAR, Chapter ~~[~~8~~]~~.
3. FSAR, Chapter ~~[~~6~~]~~.
4. FSAR, Chapter ~~[~~15~~]~~.
5. IEEE-485-~~[~~1983~~], June 1983~~.

U
U
2
1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.6 BASES, BATTERY PARAMETERS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes have been made to reflect changes made to other Specifications.

4. These battery design values have been deleted because they are more specific than necessary and are not required to provide sufficient background for this Specification.

5. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

6. Changes have been made to reflect changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.8.6, BATTERY PARAMETERS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

ITS 3.8.7, Inverters - Operating

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ELECTRICAL POWER SYSTEMS

3/4.8.2 ~~ONSITE POWER DISTRIBUTION SYSTEMS~~

~~A.C. DISTRIBUTION -~~ OPERATING

INVERTERS -

Add proposed LCO 3.8.7

Add proposed LCO 3.8.7 Note

A02

L01

R1

LIMITING CONDITION FOR OPERATION

LCO 3.8.7

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generator sets:

4160	volt Emergency Bus	1A3
4160	volt Emergency Bus	1B3
480	volt Emergency Bus	1A2
480	volt Emergency Bus	1B2
480	volt Emergency MCC Busses	1A5, 1A6, 1A7
480	volt Emergency MCC Busses	1B5, 1B6, 1B7
120	volt A.C. Instrument Bus	1MA
120	volt A.C. Instrument Bus	1MB
120	volt A.C. Instrument Bus	1MC
120	volt A.C. Instrument Bus	1MD

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

See ITS 3.8.9

ACTION:

NOTE

Enter applicable ACTIONS of LCO 3.8.2.3, “D.C. Distribution - Operating,” for DC trains made inoperable by inoperable AC distribution system.

a. With less than the above complement of A.C. emergency busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. Instrument Bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus: (1) re-energize the A.C. Instrument Bus within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) re-energize the A.C. Instrument Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Add proposed Condition A and Required Action A.1

ACTION A

ACTION B

ACTION B.2 NOTE

A02

See ITS 3.8.9

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1 4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators in accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing indicated power availability~~.

correct inverter voltage, frequency, and alignment to 120V AC instrument buses

M03

ELECTRICAL POWER SYSTEMS

INVERTERS -

3/4.8.3 ~~ONSITE POWER DISTRIBUTION~~

OPERATING

LIMITING CONDITION FOR OPERATION

Add proposed LCO 3.8.7

A02

LCO 3.8.7

3.8.3.1 The following electrical busses shall be energized in the specified manner with both tie breakers open between redundant busses and between St. Lucie Unit 1 and Unit 2.

a. Train A A.C. Emergency Busses consisting of:

1.	4160 volt Emergency Bus	# 2A3
2.	480 volt Emergency Bus	# 2A2
3.	480 volt Emergency Bus	# 2A5
4.	480 volt MCC Emergency Bus	# 2A5
5.	480 volt MCC Emergency Bus	# 2A6
6.	480 volt MCC Emergency Bus	# 2A7
7.	480 volt MCC Emergency Bus	# 2A8
8.	480 volt MCC Emergency Bus	# 2A9

b. Train B A.C. Emergency Busses consisting of:

1.	4160 volt Emergency Bus	# 2B3
2.	480 volt Emergency Bus	# 2B2
3.	480 volt Emergency Bus	# 2B5
4.	480 volt MCC Emergency Bus	# 2B5
5.	480 volt MCC Emergency Bus	# 2B6
6.	480 volt MCC Emergency Bus	# 2B7
7.	480 volt MCC Emergency Bus	# 2B8
8.	480 volt MCC Emergency Bus	# 2B9

See ITS 3.8.9

c. 120 volt A.C. Instrument Bus # 2MA energized from its associated inverter connected to D.C. Bus # 2A*.

d. 120 volt A.C. Instrument Bus # 2MB energized from its associated inverter connected to D.C. Bus # 2B*.

e. 120 volt A.C. Instrument Bus # 2MC energized from its associated inverter connected to D.C. Bus # 2A*.

f. 120 volt A.C. Instrument Bus # 2MD energized from its associated inverter connected to D.C. Bus # 2B*.

See ITS 3.8.9

g. 125 volt D.C. Bus # 2A energized from Battery Bank # 2A.

h. 125 volt D.C. Bus # 2B energized from Battery Bank # 2B.

See ITS 3.8.9

Applicability **APPLICABILITY:** MODES 1, 2, 3, and 4.

associated AC instrument

to

* Two inverters may be disconnected from their D.C. Bus for up to 24 hours~~, as necessary, for the purpose of~~ performing an equalizing charge on their associated battery bank provided (1) their ~~vital~~ busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. Bus.

LCO 3.8.7 NOTE

OPERABLE

from their Class 1E constant voltage source transformer

M02

ELECTRICAL POWER SYSTEMS

ACTION:

See ITS 3.8.9

NOTE

Enter applicable ACTIONS of LCO 3.8.2.1, “D.C. Sources - Operating,” for DC trains made inoperable by inoperable AC distribution system.

a. With one of the required trains of A.C. Emergency busses not fully energized, re-energize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. Instrument Bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus: (1) re-energize the A.C. Instrument Bus within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) re-energize the A.C. Instrument Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

A02

R1

Add proposed Condition A and Required Action A.1

ACTION A

ACTION B

ACTION B.2 NOTE

c. With one D.C. Bus not energized from its associated Battery Bank, re-energize the D.C. Bus from its associated Battery Bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

See ITS 3.8.9

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

4.8.3.1 The specified busses shall be determined energized ~~in the required manner~~ in accordance with the Surveillance Frequency Control Program by verify~~ing correct breaker alignment and indicated voltage on the busses~~.

correct inverter voltage, frequency, and alignment to 120V AC instrument buses

M03

R1

DISCUSSION OF CHANGES
ITS 3.8.7, INVERTERS - OPERATING

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS does not explicitly have a specific requirement for Train A and Train B inverters to be OPERABLE or Required Actions for inoperable inverters. ITS LCO 3.8.7 requires Train A and Train B inverters to be OPERABLE. Functionality of inverters is currently necessary to meet Unit 1 CTS 3.8.2.1 Action b and Unit 2 CTS 3.8.3.1 Action b. ITS 3.8.7 ACTION A covers the condition of one inoperable Train A or Train B inverter. ITS 3.8.7 ACTION A requires the restoration of the inoperable inverter to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program and provides a Note to direct entry into ITS 3.8.9, “Distributions Systems – Operating,” ACTIONS with any 120V AC instrument bus deenergized. ITS 3.8.7 ACTION B covers the inability of Condition A to be met and requires the plant be in MODE 3 within 6 hours and MODE 4 within 12 hours with a Note stating that LCO 3.0.4.a is not applicable when entering MODE 4. These actions are equivalent to the actions specified in Unit 1 CTS 3.8.2.1 Action b and Unit 2 CTS 3.8.3.1 Action b. R1 R1 R1

These changes are designated as administrative changes and are acceptable because they provide clarity regarding the current requirements associated with the inverters and do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Not used. R1

M02 **Unit 2 only:** Unit 2 CTS 3.8.3.1 Note * provides an allowance for two inverters to be disconnected from their DC source for up to 24 hours for performing an equalizing charge on their associated battery bank if certain provisions are maintained. The provisions require that the vital busses are energized and vital busses associated with the other battery bank are energized from their associated OPERABLE inverters. ITS 3.8.7 Note provides a similar allowance and specifies that the affected 120V instrument busses are energized from their Class 1E constant voltage source transformer. This changes the CTS by specifying that the inverters that are disconnected from their DC source for the performance of an equalizing charge on the associated battery bank shall be energized from their respective Class 1E constant voltage source transformer.

The purpose of Unit 2 CTS 3.8.3.1 is to ensure stable and reliable AC electrical power for the system instrumentation required for a safe reactor shut down. CTS 3.8.3.1 Note * provides an allowance to disconnect the inverters from their DC source to preclude damage to the inverters when the associated battery bank is

undergoing an equalizing charge. Requiring the AC instrument bus to be energized from its respective inverter powered from the Class 1E constant voltage source transformer ensures the AC instrument bus continues to be powered from a stable power source to minimize perturbations on the AC instrumentation bus. This change is designated as more restrictive because ITS requires the instrumentation busses associated with inverters disconnected from their DC source during an equalizing charge on the associated battery bank to be energized from a specific source of power.

M03 Unit 1 CTS 4.8.2.1 and Unit 2 CTS 4.8.3.1 do not specifically apply to inverter operability but requires, in part, that specified buses be determined OPERABLE by verifying indicated power availability for Unit 1 and correct breaker alignment and indicated voltage on the buses for Unit 2. ITS SR 3.8.7.1 requires the verification of correct inverter voltage, frequency, and alignment to required AC vital buses. This changes the CTS by requiring the specific verification of inverter voltage and frequency. R1 R1

The purpose of this change is to ensure the instrumentation channels are provided with the proper voltage and frequency from the AC instrument bus when powered by the associated inverter. This change is acceptable because the Surveillance will verify OPERABILITY of the inverters. Proper voltage and frequency are supplied to the instrumentation channels that provide inputs to the vital instrument systems (e.g., Reactor Protective System and Engineered Safety Features Actuation System). This change is designated as more restrictive because the ITS requires verification of the correct voltage and frequency, where the CTS does not provide explicit requirements for the inverter. R1

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 **Unit 1 only:** (*Category 1 – Relaxation of LCO Requirements*) CTS 3.8.2.1 for the Onsite Power Distribution Systems lists A.C. Distribution – Operating requirements. CTS LCO 3.8.2.1 states, “The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generator sets.” The requirement specifies four 120-volt A.C. instrument buses. CTS 3.8.2.1 Action b provides actions when the A.C. instrument bus is not energized from its associated inverter or when the inverter is not connected to its associated D.C. bus. ITS LCO 3.8.7, “Inverters – Operating” requires the A and B Train inverters to be OPERABLE. A Note is added to the LCO requirements. The Note states “Two inverters may be disconnected from their associated DC bus for ≤ 24 hours to perform an equalizing charge on their associated common R1

R1

battery, provided: The associated 120V AC instrument bus(es) are energized from their Class 1E constant voltage source transformer and all other 120V AC instrument buses are energized from their associated OPERABLE inverters." This changes the CTS by allowing two inverters to be disconnected from their DC bus for up to 24 hours during a battery equalize charge without requiring entry into LCO 3.0.3.

The purpose of ITS 3.8.7 LCO Note is to allow required maintenance on a battery without entry into LCO 3.0.3, which would require a plant shutdown transient. This change is acceptable because the LCO requirements continue to ensure that the associated 120V AC instrument buses are energized and avoid a condition that would require an unnecessary plant transient. The allowance provides 24 hours to perform an equalize change on an associated battery without requiring entry into a condition for two inoperable inverters (i.e., LCO 3.0.3). In this situation, the Note provides an exception to ITS LCO 3.8.7, which requires Train A and B inverters to be OPERABLE from their associated common DC battery. A battery may periodically require an equalize charge to ensure its continued OPERABILITY. The proposed allowance provides a limited time for the required maintenance on the battery and precludes possible damage to the associated inverters. The ITS requirement is consistent with the allowance specified in the Unit 2 CTS and is consistent with the ISTS wording for this requirement. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

3.8.2.1 DOC A02

LCO 3.8.7 The ~~required~~ Train A and Train B inverters shall be OPERABLE. (1)

T

DOC L01

----------NOTE----------
~~[~~ ~~[One/~~two] inverter~~[s]~~ may be disconnected from ~~[its/~~their~~]~~ associated DC bus for ≤ 24 hours to perform an equalizing charge on ~~[its/~~their~~]~~ associated ~~[~~common~~]~~ battery, provided:

120V AC instrument

a. The associated ~~AC vital~~ bus(es) ~~[is/~~are~~]~~ energized from ~~[its/~~their~~]~~ ~~[~~Class 1E constant voltage source transformer~~s] [inverter using internal AC source]~~ and (2) (3)

120V AC instrument

b. All other ~~AC vital~~ buses are energized from their associated OPERABLE inverters.~~]~~ (2)

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC A02 A. One ~~[required]~~ inverter inoperable. (3)	A.1 ----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems – Operating," with any ~~vital~~ bus de-energized. ---------- (2) 120V AC instrument	Restore inverter to OPERABLE status. 24 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~ (3) R1

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC A02 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.2.1 DOC M03 SR 3.8.7.1	Verify correct inverter voltage, ~~[~~frequency,~~]~~ and alignment to ~~required AC vital~~ buses. 120V AC instrument	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (1) (2) (3)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

3.8.3.1 DOC A02

LCO 3.8.7 The ~~required~~ Train A and Train B inverters shall be OPERABLE. (1)

T

----------NOTE----------

DOC M02

~~[~~ ~~[One/~~two] inverter~~[~~s~~]~~ may be disconnected from ~~[its/~~their~~]~~ associated DC bus for ≤ 24 hours to perform an equalizing charge on ~~[its/~~their~~]~~ associated ~~[~~common~~]~~ battery, provided:

120V AC instrument

a. The associated ~~AC vital~~ bus(es) ~~[is/~~are~~]~~ energized from ~~[its/~~their~~]~~ ~~[~~Class 1E constant voltage source transformer~~s] [inverter using internal AC source]~~ and (2) (3)

120V AC instrument

b. All other ~~AC vital~~ buses are energized from their associated OPERABLE inverters.~~]~~ (2)

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC A02 A. One ~~[required]~~ inverter inoperable.	A.1 ----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems – Operating," with any ~~vital~~ bus de-energized. ---------- (3)	
120V AC instrument	Restore inverter to OPERABLE status. (2)	24 hours ~~[~~OR In accordance with the Risk Informed Completion Time Program~~]~~ (3) | R1

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC A02 B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.3.1 DOC M03 SR 3.8.7.1	Verify correct inverter voltage, ~~[~~frequency,~~]~~ and alignment to ~~required AC vital~~ buses. 120V AC instrument	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~

1 2 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.7, INVERTERS - OPERATING

1. The word “required” has been deleted from the LCO and SR 3.8.7.1 since all Train A and Train B inverters are required.

2. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the AC ~~vital~~ buses because of the stability and reliability they achieve. The function of ~~the~~ inverter is to provide AC electrical power to the ~~vital~~ bus~~es~~. The inverters ~~can be~~ powered from ~~an internal AC source/rectifier or from~~ the station batter~~y~~. The ~~station battery~~ provide~~s~~ an uninterruptible power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in the FSAR, Chapter ~~[~~8~~]~~ (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 2) and Chapter ~~[~~15~~]~~ (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC ~~vital~~ buses OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power and

b. A worst case single failure.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

BASES

LCO (continued)

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The four inverters ~~[~~(two per train)~~]~~ ensure an uninterruptible supply of 120 V AC electrical power to the AC instrument ~~vital~~ buses even if the 4.16 kV safety buses are de-energized.

OPERABLE inverters require the associated instrument ~~vital~~ bus to be powered by the inverter with output voltage and frequency within tolerances, and power input to the inverter from a ~~[~~125 VDC~~]~~ station battery. ~~Alternatively, power supply may be from an internal AC source via rectifier as long as the station battery is available as the uninterruptible power supply~~.

This LCO is modified by a Note that allows ~~[one/~~two~~]~~ inverters to be disconnected from a ~~[~~common~~]~~ battery for ≤ 24 hours, if the instrument ~~vital~~ bus(es) are ~~is~~ powered from a ~~[~~Class 1E constant voltage transformer ~~or inverter using internal AC source]~~ during the period and all other inverters are OPERABLE ~~operable~~. This allows an equalizing charge to be placed on one battery. If the inverter(s) were not disconnected, the resulting voltage condition might damage the inverter(s). These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 24 hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected AC instrument ~~vital~~ bus while taking into consideration the time required to perform an equalizing charge on the battery bank.

The intent of this Note is to limit the number of inverters that may be disconnected. Only those inverters associated with the single battery undergoing an equalizing charge may be disconnected. All other inverters must be aligned to their associated batteries, regardless of the number of inverters or unit design.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."

BASES

ACTIONS

A.1

Train A or Train B | 120 V | instrument

With a ~~required~~ inverter inoperable, its associated AC ~~vital~~ bus becomes inoperable ~~until~~ it is [manually] re-energized from its [Class 1E constant voltage source transformer ~~or inverter using internal AC source~~].

unless

LCO 3.8.9, "Distribution Systems – Operating," addresses this action however, pursuant to LCO 3.0.6, this action would not have to be entered even if the 120 VAC instrument bus were de-energized.

Required Action A.1 is modified by a Note, which states to enter the applicable conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," when Condition A is entered with ~~one AC vital~~ bus de-energized. This ensures the ~~vital~~ bus is re-energized within 2 hours.

120 VAC instrument | any 120 VAC instrument

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.] The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the ~~AC vital~~ bus is powered from its constant voltage ~~source~~, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the ~~AC vital~~ buses is the preferred source for powering instrumentation trip setpoint devices.

120 VAC instrument | Class 1E | 120 VAC instrument | transformer

R1

B.1 and B.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

1. ~~[LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

2. ~~[LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

Train A or Train B inverter

If the inoperable ~~devices or components~~ cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.

BASES

ACTIONS (continued)

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

120 VAC instrument

associated

120 VAC instrument

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and ~~AC vital~~ buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the ~~AC vital~~ buses. ~~[The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

3

2

BASES

REFERENCES

1. FSAR, Chapter [8].
2. FSAR, Chapter [6].
3. FSAR, Chapter [14].
4. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the AC ~~vital~~ buses because of the stability and reliability they achieve. The function of ~~the~~ inverter is to provide AC electrical power to the ~~vital~~ bus~~es~~. The inverters ~~can be~~ powered from ~~an internal AC source/rectifier or from~~ the station batter~~y~~. The ~~station battery~~ provide~~s~~ an uninterruptible power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in the FSAR, Chapter ~~[~~8~~]~~ (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 2) and Chapter ~~[~~15~~]~~ (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC ~~vital~~ buses OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power and

b. A worst case single failure.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

BASES

LCO (continued)

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The four inverters ~~[~~(two per train)~~]~~ ensure an uninterruptible supply of 120 V AC electrical power to the AC instrument ~~vital~~ buses even if the 4.16 kV safety buses are de-energized.

OPERABLE inverters require the associated instrument ~~vital~~ bus to be powered by the inverter with output voltage and frequency within tolerances, and power input to the inverter from a ~~[~~125 VDC~~]~~ station battery. ~~Alternatively, power supply may be from an internal AC source via rectifier as long as the station battery is available as the uninterruptible power supply~~.

This LCO is modified by a Note that allows ~~[one/~~two~~]~~ inverters to be disconnected from a ~~[~~common~~]~~ battery for ≤ 24 hours, if the instrument ~~vital~~ bus(es) are ~~is~~ powered from a ~~[~~Class 1E constant voltage transformer ~~or inverter using internal AC source]~~ during the period and all other inverters are OPERABLE ~~operable~~. This allows an equalizing charge to be placed on one battery. If the inverter(s) were not disconnected, the resulting voltage condition might damage the inverter(s). These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 24 hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected AC instrument ~~vital~~ bus while taking into consideration the time required to perform an equalizing charge on the battery bank.

The intent of this Note is to limit the number of inverters that may be disconnected. Only those inverters associated with the single battery undergoing an equalizing charge may be disconnected. All other inverters must be aligned to their associated batteries, regardless of the number of inverters or unit design.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."

BASES

ACTIONS

A.1

Train A or Train B
120 V
instrument
unless
With a ~~required~~ inverter inoperable, its associated AC ~~vital~~ bus becomes inoperable ~~until~~ it is [manually] re-energized from its [Class 1E constant voltage source transformer ~~or inverter using internal AC source~~]. 1 2 1

LCO 3.8.9, "Distribution Systems – Operating," addresses this action however, pursuant to LCO 3.0.6, this action would not have to be entered even if the 120 VAC instrument bus were de-energized.

Required Action A.1 is modified by a Note, which states to enter the applicable conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," when Condition A is entered with ~~one AC vital~~ bus de-energized. This ensures the ~~vital~~ bus is re-energized within 2 hours. 1

120 VAC instrument
any 120 VAC instrument

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. [Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.] The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. 2 R1

120 VAC instrument
Class 1E
120 VAC instrument
When the ~~AC vital~~ bus is powered from its constant voltage ~~source~~, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the ~~AC vital~~ buses is the preferred source for powering instrumentation trip setpoint devices. 1

transformer

B.1 and B.2

~~REVIEWER'S NOTE~~

~~Adoption of a MODE 4 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision [4F].~~

~~2. [LICENSEE] will follow the guidance established in Revision 2 of WCAP-16364-NP, "Implementation Guidance for Risk Informed Modification to Selected Required Action End States at Combustion Engineering NSSS Plants (TSTF-422)," Westinghouse, May 2010.~~

3

Train A or Train B inverter

If the inoperable ~~devices or components~~ cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. 1

BASES

ACTIONS (continued)

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

120V AC instrument

associated

120 VAC instrument

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and ~~AC vital~~ buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the ~~AC vital~~ buses. ~~[The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.~~

1

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~

3

2

BASES

REFERENCES

1. FSAR, Chapter [8].
2. FSAR, Chapter [6].
3. FSAR, Chapter [14].
4. CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001.

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.7 BASES, INVERTERS - OPERATING

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

4. Corrections have been made consistent with the Writer's Guide for the Improved Technical Specifications, TSTF-GG-05-01.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.8.7, INVERTERS - OPERATING

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 8

ITS 3.8.8, Inverters - Shutdown

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ELECTRICAL POWER SYSTEMS

ELECTRICAL POWER SYSTEMS

INVERTERS -

~~ONSITE POWER DISTRIBUTION~~

SHUTDOWN

LIMITING CONDITION FOR OPERATION

Add proposed LCO 3.8.8

A02

LCO 3.8.8

3.8.3.2 As a minimum, the following electrical busses shall be energized and in the specified manner:

a. One train of A.C. emergency busses consisting of one 4160 volt and two 480 volt A.C. emergency busses.

See ITS 3.8.10

b. Two 120 volt A.C. Instrument Busses energized from their associated inverters connected to their respective D.C. busses.

See ITS 3.8.10

c. One 125 volt D.C. bus energized from its associated battery bank.

Applicability

APPLICABILITY: MODES 5 and 6~~.~~ ,

During movement of irradiated fuel assemblies.

M02

ACTION:

Add proposed ACTIONS NOTE

Add proposed Required Action A.1

L01

L02

ACTION A

With any of the above required electrical busses not energized in the required manner, immediately suspend ~~all operations involving CORE ALTERATIONS,~~ operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, or movement of irradiated fuel assemblies, initiate corrective action to energize the required electrical busses ~~in the specified manner as soon as possible, and within 8 hours depressurize and vent the RCS through a 3.58 square inch vent~~.

See ITS 3.8.10

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

4.8.3.2 ~~The specified busses shall be determined energized in the required manner~~ in accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing~~ correct ~~breaker alignment and indicated voltage on the~~ busses.

M03

inverter voltage, frequency, and alignments to required 120V ac instrument

DISCUSSION OF CHANGES
ITS 3.8.8, INVERTERS - SHUTDOWN

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 2 only:** CTS does not have a specific requirement for inverters to be OPERABLE or Required Actions for inoperable required inverters. However, CTS 3.8.3.2, “Onsite Power Distribution – Shutdown” requires 120V AC instrument busses to be powered from their associated inverters, requiring functionality of the inverters. A new ITS LCO 3.8.8, specifically requires “required” inverters to be OPERABLE to support the 120V AC electrical distribution subsystems required by LCO 3.8.10 when in MODE 5, MODE 6, and during movement of irradiated fuel assemblies. This changes the CTS by adding specific operability requirements for inverters associated with the 120V AC instrument busses necessary to support LCO 3.8.10 requirements.

These changes are designated as administrative changes as they are a clarification of the intent of CTS 3.8.3.2 and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS does not have a requirement for inverters to be OPERABLE, Required Actions or Surveillance Requirements. ITS LCO 3.8.8 requires inverters to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies to support the 120V AC electrical distribution subsystem required by ITS LCO 3.8.10, “Distribution Systems - Shutdown.”. In addition, ITS 3.8.8 provides the necessary Required Actions and Surveillance Requirement that support inverter OPERABILITY. PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per CTS 6.8.4.o. Therefore, SR 3.8.8.1 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 7 days consistent with the ISTS. This changes the CTS by incorporating the requirements of ITS 3.8.8.

The safety related function of required inverters is to provide an uninterruptible power supply for the 120 VAC instrument buses. This change is acceptable because the safety analyses assume that the loads supported by the 120 VAC instrument buses have an uninterruptible supply of AC electrical power. This change is designated as more restrictive because it adds new requirements to the CTS.

M02 **Unit 2 only:** CTS does not have a specific requirement for inverters to be OPERABLE or Applicability requirements. See DOC A02 discussion. ITS LCO 3.8.8 is applicable in MODE 5, MODE 6, and during movement of irradiated fuel assemblies. In addition, a Note modifies the ACTIONS to state that LCO 3.0.3 is not applicable. This changes the CTS by adding an Applicability of "during movement of irradiated fuel assemblies" and adds the Note to the ACTIONS stating that LCO 3.0.3 is not applicable.

This change is acceptable because the proposed requirements are necessary to ensure the inverters are OPERABLE to support equipment required to be OPERABLE during movement of irradiated fuel assemblies. Movement of fuel normally occurs during MODES 5 and 6, however, it can also occur outside of containment in other plant MODES (MODES 1, 2, 3, and 4) or other conditions (i.e., reactor defueled). This Specification is needed to ensure the appropriate distribution system requirements are specified during fuel handling and ensure that appropriate ACTIONS are taken (i.e. stop fuel movement) when the minimum electrical supply is not available. This change adds a clarification Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODES 5 or 6, LCO 3.0.3 is not applicable and would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations and the inability to suspend movement in accordance with the ITS 3.8.8 Required Actions would not be sufficient reason to require a reactor shutdown. This Note has been added for clarification and is necessary since defaulting to LCO 3.0.3 would require the reactor to be shutdown but would not require suspension of activities with a potential for releasing radioactive materials. This change is designated as more restrictive because the ITS requires the equipment to be OPERABLE during movement of irradiated fuel assemblies both inside and outside of the containment in conditions other than MODES 5 and 6.

M03 **Unit 2 only:** Unit 2 CTS 4.8.3.2 does not specifically apply to inverter operability but requires, in part, that specified 120V AC instrument buses be determined OPERABLE by verifying correct breaker alignment. ITS SR 3.8.8.1 requires the verification of correct inverter voltage, frequency, and alignment to required AC vital buses. This changes the CTS by requiring the specific verification of inverter voltage output and frequency.

The purpose of this requirement is to ensure the required instrumentation channels are provided with the proper voltage and frequency from the AC instrument bus when powered by the associated inverter. This change is acceptable because the Surveillance will verify OPERABILITY of the required inverters with the proper voltage and frequency supplied to the instrumentation channels that provide inputs to the vital instrument systems (e.g., Reactor Protective System and Engineered Safety Features Actuation System). This change is designated as more restrictive because the ITS requires verification of the correct voltage and frequency, where the CTS does not provide explicit requirements for the inverter.

DISCUSSION OF CHANGES
ITS 3.8.8, INVERTERS - SHUTDOWN

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* CTS does not have specific actions for inoperable required inverters. However, CTS 3.8.3.2, "Onsite Power Distribution – Shutdown" requires 120V ac instrument busses to be powered from the associated inverter which infers the CTS 3.8.3.2 actions also apply to the inverters. CTS 3.8.3.2 in part states "With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity changes that could result in loss of required SHUTDOWN MARGIN or boron concentration, or movement of irradiated fuel." For one or more required inverters inoperable, ITS 3.8.8 Action A.1 states "Declare affected required feature(s) inoperable." This must be performed immediately, or other specific Required Actions must be followed. This changes the CTS by allowing the affected required feature(s) with an inoperable inverter to be declared inoperable instead of suspending the specified activities.

The purpose of ITS Required Action A.1 is to provide an alternative to stopping movement of irradiated fuel assemblies and positive reactivity changes with required inverters inoperable. This change is acceptable because the Required Actions of the required supported features are also acceptable remedial actions in response to degraded conditions. These actions also minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a design basis accident occurring during the repair period. The addition of the allowance provides for declaring affected features with electrical power inverters inoperable instead of immediately suspending movement of irradiated fuel assemblies and stopping positive reactivity changes. The ITS requirement is consistent with the ISTS wording for this requirement. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* CTS does not have specific actions for inoperable required inverters. However, CTS 3.8.3.2, "Onsite Power Distribution – Shutdown" requires 120V ac instrument busses to be powered from the associated inverter which infers the CTS 3.8.3.2 actions also apply to the inverters. CTS 3.8.3.2 ACTION states, in part, "With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS. ITS 3.8.8

ACTIONS do not include the requirement to suspend CORE ALTERATIONS. This changes the CTS by removing the Required Action to suspend all operations involving CORE ALTERATIONS.

The purpose of the CTS 3.8.3.2 ACTION is to minimize the possibility of an event that may require the electrical power source to mitigate the consequences of a design basis event. CORE ALTERATIONS is defined in CTS 1.9, in part, as "the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity, within the reactor vessel with the head removed and fuel in the vessel." CORE ALTERATIONS only occur when the reactor vessel head is removed - it only applies in MODE 6. There is only one accident considered during MODE 6 that involves a Core Alteration - a fuel handling accident. According to UFSAR Section 15.7.4.1.2, a fuel handling accident is initiated by the dropping of an irradiated fuel assembly either in the containment or in the fuel handling building. Suspension of CORE ALTERATIONS, except for suspension of movement of irradiated fuel, will not prevent or impair the mitigation of a fuel handling accident. ITS 3.8.8 retains the requirement to suspend movement of irradiated fuel assemblies as Required Action A.2.1 (for one or more inoperable inverters). R1

Deleting the reference to “CORE ALTERATIONS” is acceptable because the only CORE ALTERATION analyzed in the safety analysis that is potentially affected by an inoperable inverter is covered by the ITS Required Actions. Additionally, CORE ALTERATIONS that may be capable of damaging a fuel assembly prior to the 72 hour decay period is improbable because of the physical time required to perform plant shutdown, cooldown, depressurize the Reactor Coolant System (RCS), and the additional operations required to expose irradiated fuel in the reactor vessel (e.g., containment entry, removal of vessel head, removal of vessel internals, etc.) is greater than 72 hours. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

DOC M01 LCO 3.8.8

~~[Inverter(s) shall be OPERABLE to support the onsite Class 1E AC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown."]~~

Required

~~[One]~~ inverter~~[s]~~ shall be OPERABLE.~~]~~

to support the 120V AC electrical distribution subsystem required by LCO 3.8.10, "Distribution Systems – Shutdown."

2

~~----------REVIEWER'S NOTE----------~~
~~This second option above applies for plants having a pre-ITS licensing basis (CTS) for electrical power requirements during shutdown conditions that required only [one] inverter to be OPERABLE. The "[or more]" optional wording in Condition A is also eliminated for this case. The first option above is adopted for plants that have a CTS requiring the same level of DC electrical power subsystem/inverter support as is required for power operating conditions.~~
~~----------~~

3

DOC M01 APPLICABILITY: MODES 5 and 6,
During movement of ~~[recently]~~ irradiated fuel assemblies.

2

DOC M01 ACTIONS

----------NOTE----------
LCO 3.0.3 is not applicable

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~[~~or more~~]~~ ~~[~~required~~]~~ inverter~~[s]~~ inoperable.	A.1 Declare affected required feature(s) inoperable. OR A.2.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies. AND	Immediately Immediately

2

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. AND A.2.3 Initiate action to restore required inverters to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
DOC M01	SR 3.8.8.1	Verify correct inverter voltage, ~~[~~frequency,~~]~~ and alignments to required ~~AC vital~~ buses. 120V AC instrument	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~	1 2

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

3.8.3.2
DOC A02

LCO 3.8.8

~~[Inverter(s) shall be OPERABLE to support the onsite Class 1E AC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown."]~~

Required

~~[One]~~ inverter~~[s]~~ shall be OPERABLE.~~]~~

to support the 120V AC electrical distribution subsystem required by LCO 3.8.10, "Distribution Systems – Shutdown."

2

~~----------REVIEWER'S NOTE----------~~
~~This second option above applies for plants having a pre-ITS licensing basis (CTS) for electrical power requirements during shutdown conditions that required only [one] inverter to be OPERABLE. The "[or more]" optional wording in Condition A is also eliminated for this case. The first option above is adopted for plants that have a CTS requiring the same level of DC electrical power subsystem/inverter support as is required for power operating conditions.~~
~~----------~~

3

Applicability
DOC M02

APPLICABILITY: MODES 5 and 6,
During movement of ~~[recently]~~ irradiated fuel assemblies.

2

Action

ACTIONS

----------NOTE----------
DOC M02
LCO 3.0.3 is not applicable

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01 A. One ~~[~~or more~~]~~ ~~[~~required~~]~~ inverter~~[s]~~ inoperable.	A.1 Declare affected required feature(s) inoperable. OR A.2.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies. AND	Immediately Immediately

2

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. AND A.2.3 Initiate action to restore required inverters to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.8.3.2 DOC M03	SR 3.8.8.1	Verify correct inverter voltage, ~~[~~frequency,~~]~~ and alignments to required ~~AC vital~~ buses. 120V AC instrument	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (1) (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.8, INVERTERS - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES

U

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 1) and Chapter ~~[~~15~~]~~ (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. 1 2

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the ~~minimum~~ inverter required to each ~~AC vital~~ 120 VAC instrument bus during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that: 1 5

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident ~~[involving handling recently irradiated fuel. Due to radioactive decay, the inverters are only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~. 2

In general, when the unit is shut down, the Technical Specification requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many DBAs that are analyzed in MODES ~~[~~1, 2, 3, and 4~~]~~ have no specific analyses in MODES ~~[~~5 and 6~~]~~ because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being 2

BASES

APPLICABLE SAFETY ANALYSES (continued)

significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

electrical power

1

LCO

The inverter~~[s]~~ ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The ~~battery powered~~ inverter~~[s]~~ provide~~[s]~~ uninterruptible supply of AC electrical power to the ~~AC vital~~ bus~~[es] even if the 4.16 kV safety buses are de-energized~~. OPERABILITY of the inverter~~[s]~~ requires ~~that the vital bus be powered by the inverter~~. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents ~~[involving handling recently irradiated fuel]~~).

2

120 VAC instrument

the associated 120 VAC instrument bus to be powered by the inverter with the output voltage and frequency within tolerances, and power input from a 125 VDC station battery.

2 1

2

APPLICABILITY

The inverter~~[s]~~ required to be OPERABLE in MODES 5 and 6 during movement of ~~[recently]~~ irradiated fuel assemblies provide assurance that:

, and

5

2

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core,

b. Systems needed to mitigate a fuel handling accident ~~[involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~ are available,

2

BASES

APPLICABILITY (continued)

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

3

4

A.1, A.2.1, A.2.2, and A.2.3

one or more inverters

~~[~~If ~~two trains~~ are required by LCO 3.8.10, "Distribution Systems - Shutdown," the remaining OPERABLE inverters may be capable of supporting sufficient required features to allow continuation of ~~[recently]~~ irradiated fuel movement, operations with a potential for draining the reactor vessel, and operations with a potential for positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6).~~]~~ Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive ~~MTC~~ must also be evaluated to ensure they do not result in a loss of required SDM. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of ~~[recently]~~ irradiated fuel assemblies, and operations involving positive reactivity additions).

2

moderator temperature coefficient

4

2

BASES

ACTIONS (continued)

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverter[s] and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems. 2

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer. 1

the associated

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC vital buses. [The 7-day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions. 1 2

120 VAC instrument

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement. 3
] 2

REFERENCES

1. FSAR, Chapter [6]. 2 1
2. FSAR, Chapter [15].

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES

U

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 1) and Chapter ~~[~~15~~]~~ (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. 1 2

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the ~~minimum~~ inverter required to each ~~AC vital~~ 120 VAC instrument bus during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that: 1 5

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident ~~[involving handling recently irradiated fuel. Due to radioactive decay, the inverters are only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~. 2

In general, when the unit is shut down, the Technical Specification requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many DBAs that are analyzed in MODES ~~[~~1, 2, 3, and 4~~]~~ have no specific analyses in MODES ~~[~~5 and 6~~]~~ because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being 2

BASES

APPLICABLE SAFETY ANALYSES (continued)

significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The inverters were previously identified as part of the electrical power distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). (1)

LCO

The inverter~~[s]~~ ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The ~~battery powered~~ inverter~~[s]~~ provide~~[s]~~ uninterruptible supply of AC electrical power to the ~~AC vital~~ bus~~[es] even if the 4.16 kV safety buses are de-energized~~. OPERABILITY of the inverter~~[s]~~ requires ~~that the vital bus be powered by the inverter~~. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents ~~[involving handling recently irradiated fuel]~~). (2) (2) (1) (2)

120 VAC instrument

the associated 120 VAC instrument bus to be powered by the inverter with the output voltage and frequency within tolerances, and power input from a 125 VDC station battery.

APPLICABILITY

The inverter~~[s]~~ required to be OPERABLE in MODES 5 and 6 during movement of ~~[recently]~~ irradiated fuel assemblies provide assurance that: (, and) (5) (2)

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core,

b. Systems needed to mitigate a fuel handling accident ~~[involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~ are available, (2)

BASES

APPLICABILITY (continued)

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

3

4

A.1, A.2.1, A.2.2, and A.2.3

one or more inverters

~~[~~If ~~two trains~~ are required by LCO 3.8.10, "Distribution Systems - Shutdown," the remaining OPERABLE inverters may be capable of supporting sufficient required features to allow continuation of ~~[recently]~~ irradiated fuel movement, operations with a potential for draining the reactor vessel, and operations with a potential for positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6).~~]~~ Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive ~~MTC~~ must also be evaluated to ensure they do not result in a loss of required SDM. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of ~~[recently]~~ irradiated fuel assemblies, and operations involving positive reactivity additions).

2

moderator temperature coefficient

4

2

1

BASES

ACTIONS (continued)

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverter[s] and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems. 2

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer. 1

the associated

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC vital buses. [The 7-day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions. 1 2

120 VAC instrument

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REVIEWER'S NOTE
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.] 3 2

REFERENCES

1. FSAR, Chapter [6]. 2 1
2. FSAR, Chapter [15].

U

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.8 BASES, INVERTERS - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

4. Changes are made to correct an editorial error.

5. Changes are made to reflect changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.8.8, INVERTERS - SHUTDOWN

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 9

ITS 3.8.9, Distribution Systems - Operating

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

Train A and Train B AC, DC, and AC instrument bus electrical power subsystems

LIMITING CONDITION FOR OPERATION

LCO 3.8.9

3.8.2.1 ~~The following A.C. electrical busses~~ shall be OPERABLE ~~and energized from sources of power other than the diesel generator sets:~~ LA02 L01

~~4160~~	~~volt Emergency Bus~~	~~1A3~~
~~4160~~	~~volt Emergency Bus~~	~~1B3~~
~~480~~	~~volt Emergency Bus~~	~~1A2~~
~~480~~	~~volt Emergency Bus~~	~~1B2~~
~~480~~	~~volt Emergency MCC Busses~~	~~1A5, 1A6, 1A7~~
~~480~~	~~volt Emergency MCC Busses~~	~~1B5, 1B6, 1B7~~
~~120~~	~~volt A.C. Instrument Bus~~	~~1MA~~
~~120~~	~~volt A.C. Instrument Bus~~	~~1MB~~
~~120~~	~~volt A.C. Instrument Bus~~	~~1MC~~
~~120~~	~~volt A.C. Instrument Bus~~	~~1MD~~

LA01

Applicability **APPLICABILITY:** MODES 1, 2, 3 and 4.

ACTION:

ACTION A NOTE

NOTE

4 Sources

Enter applicable ACTIONS of LCO 3.8.~~2.3~~, "D.C. ~~Distribution~~ - Operating," for DC trains made inoperable by inoperable AC distribution system.

one or more AC electrical power distribution subsystems inoperable

ACTION A a. With ~~less than the above complement of A.C. emergency busses OPERABLE~~, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION D

or more es inoperable

ACTION B b. With one A.C. Instrument Bus ~~either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus:~~ (1) re-energize the A.C. Instrument Bus within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours ~~and~~ (2) re-energize the A.C. Instrument Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN. R1

ACTION D

See ITS 3.8.7

M01

Add proposed ACTION E

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1 4.8.2.1 ~~The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators~~ in accordance with the Surveillance Frequency Control Program by verify~~ing indicated power availability~~. LA02 L01

correct breaker alignments and voltage to required AC, and AC instrument bus electrical power distribution subsystems M02

ELECTRICAL POWER SYSTEMS

SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.8.9

3.8.2.3 Train A and Train B ~~As a minimum the following~~ D.C. electrical power distribution subsystems ~~sources~~ shall be OPERABLE:

a. ~~125-volt D.C. bus No. 1A~~, 125-volt Battery bank No. 1A and a full capacity charger.

b. ~~125-volt D.C. bus No. 1B,~~ 125-volt Battery bank No. 1B and a full capacity charger.

See ITS 3.8.4

LA01

LA01

Applicability **APPLICABILITY**: MODES 1, 2, 3 and 4.

See ITS 3.8.4

ACTION:

DC electrical power distribution subsystem bus

ACTION C

ACTION D

a. With one ~~of the required~~ battery banks ~~or busses~~ inoperable, restore the inoperable battery bank or bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in ~~HOT~~ SHUTDOWN within the following ~~6~~ hours. ~~LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.~~

30

COLD

M03

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.3.2.a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

See ITS 3.8.4

Add proposed ACTION E

M01

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

4.8.2.3.1 ~~Each D.C. bus train shall be determined OPERABLE and energized~~ in accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing indicated power availability~~.

correct breaker alignments and voltage to required DC electrical power distribution subsystems

LA02

M02

4.8.2.3.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129-volts on float charge.

See ITS 3.8.4

ELECTRICAL POWER SYSTEMS

3/4.8.3 ~~ONSITE POWER~~ DISTRIBUTION

SOURCES -

OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.8.9

3.8.3.1 ~~The following electrical busses shall be energized in the specified manner with both tie breakers open between redundant busses and between St. Lucie Unit 1 and Unit 2.~~

LA02

~~a. Train A A.C. Emergency Busses consisting of:~~

~~1.~~	~~4160 volt Emergency Bus~~	~~# 2A3~~
~~2.~~	~~480 volt Emergency Bus~~	~~# 2A2~~
~~3.~~	~~480 volt Emergency Bus~~	~~# 2A5~~
~~4.~~	~~480 volt MCC Emergency Bus~~	~~# 2A5~~
~~5.~~	~~480 volt MCC Emergency Bus~~	~~# 2A6~~
~~6.~~	~~480 volt MCC Emergency Bus~~	~~# 2A7~~
~~7.~~	~~480 volt MCC Emergency Bus~~	~~# 2A8~~
~~8.~~	~~480 volt MCC Emergency Bus~~	~~# 2A9~~

~~b. Train B A.C. Emergency Busses consisting of:~~

~~1.~~	~~4160 volt Emergency Bus~~	~~# 2B3~~
~~2.~~	~~480 volt Emergency Bus~~	~~# 2B2~~
~~3.~~	~~480 volt Emergency Bus~~	~~# 2B5~~
~~4.~~	~~480 volt MCC Emergency Bus~~	~~# 2B5~~
~~5.~~	~~480 volt MCC Emergency Bus~~	~~# 2B6~~
~~6.~~	~~480 volt MCC Emergency Bus~~	~~# 2B7~~
~~7.~~	~~480 volt MCC Emergency Bus~~	~~# 2B8~~
~~8.~~	~~480 volt MCC Emergency Bus~~	~~# 2B9~~

LA01

~~c. 120 volt A.C. Instrument Bus # 2MA energized from its associated inverter connected to D.C. Bus # 2A~~*.

~~d. 120 volt A.C. Instrument Bus # 2MB energized from its associated inverter connected to D.C. Bus # 2B~~*.

~~e. 120 volt A.C. Instrument Bus # 2MC energized from its associated inverter connected to D.C. Bus # 2A~~*.

~~f. 120 volt A.C. Instrument Bus # 2MD energized from its associated inverter connected to D.C. Bus # 2B~~*.

L01 LA01

~~g. 125 volt D.C. Bus # 2A energized from Battery Bank # 2A.~~

~~h. 125 volt D.C. Bus # 2B energized from Battery Bank # 2B.~~

Applicability **APPLICABILITY:** MODES 1, 2, 3, and 4.

* Two inverters may be disconnected from their D.C. Bus for up to 24 hours, as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. Bus.

See ITS 3.8.7

ELECTRICAL POWER SYSTEMS

ACTION:

Required Action A.1 NOTE

NOTE

Enter applicable ACTIONS of LCO 3.8.~~2.1~~ 4, "D.C. Sources - Operating," for DC trains made inoperable by inoperable AC distribution system.

ACTION A
ACTION D

a. With one of the required trains of A.C. Emergency busses ~~not fully energized~~ inoperable, re-energize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION B
ACTION D

b. With one A.C. Instrument Bus ~~either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus:~~ inoperable (1) re-energize the A.C. Instrument Bus within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours ~~and~~ (2) re-energize the A.C. Instrument Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN. R1

See ITS 3.8.7

ACTION C
ACTION D

c. With one D.C. Bus ~~not energized from its associated Battery Bank, re-energize the D.C. Bus from its associated Battery Bank~~ inoperable within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

restore DC electrical power distribution subsystem to OPERABLE

Add proposed ACTION E

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

4.8.3.1 ~~The specified busses shall be determined energized in the required manner~~ in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and ~~indicated~~ voltage ~~on the busses~~.

LA02

to required DC electrical power distribution subsystems

DISCUSSION OF CHANGES
ITS 3.8.9, DISTRIBUTION SYSTEMS - OPERATING

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.8.2.1 Action a states that with less than the above complement of AC buses OPERABLE, to restore the inoperable bus to OPERABLE status within 8 hours. Unit 1 CTS 3.8.2.3 Action a states that with one required 125 VDC bus inoperable, to restore the inoperable bus to OPERABLE status within 2 hours. Unit 2 CTS 3.8.3.1 Action a states with one required train of A.C. Emergency busses not fully energized, re-energize the train within 8 hours, Action b states that with one A.C. Instrument Bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus reenergize the instrument bus within 2 hours, and Action c states that with one D.C. bus not energized from its associated Battery Bank to re-energize the D.C. Bus from its associated Battery Bank with 2 hours. However, there are no limitations to preclude a loss of function due to numerous concurrently inoperable AC and DC buses. ITS 3.8.9 ACTION E has been added, requiring entry into ITS LCO 3.0.3 if two or more inoperable electrical power distribution subsystems result in a loss of safety function. | R1 | R1

The purpose of the CTS Actions is to limit the time the unit can operate under these conditions. Unit 1 CTS 3.8.2.3 Action a specifies the compensatory actions for one inoperable DC bus. With two inoperable DC buses, Unit 1 CTS 3.8.2.3 does not provide any actions and entry into LCO 3.0.3 would be required. Unit 1 CTS 3.8.2.1 Action a is applicable to all inoperable AC buses even if there is a loss of safety function. Certain combinations of inoperable AC and DC electrical power distribution subsystems may result in a loss of safety function (e.g., an inoperable Train A AC electrical power distribution subsystem in combination with an inoperable Train B DC electrical power distribution subsystem). Unit 2 CTS 3.8.3.1 specifies the compensatory actions for one inoperable A.C. Emergency bus train inoperable, one A.C. Instrument Bus inoperable, or one D.C. bus inoperable but does not provide specific actions for conditions that result in a loss of safety function. ITS 3.8.9 includes ACTION E, which requires immediate entry into LCO 3.0.3 if two or more inoperable electrical power distribution subsystems result in a loss of safety function. ITS 3.8.9 Required Action E.1 preserves the intent of ITS LCO 3.0.3 and reflects an additional restriction on plant operation. This change is designated as more restrictive because an explicit action has been added which requires entry into LCO 3.0.3 with any combination of AC and/or DC buses inoperable that result in a loss of safety function. | R1 | R1

M02 **Unit 1 only:** CTS 4.8.2.1 requires, in part, the specified A.C. busses shall be determined OPERABLE by verifying "indicated power availability" and CTS 4.8.2.3.1 states each D.C. bus train shall be determined OPERABLE by verifying "indicated power availability." ITS SR 3.8.9.1 requires the verification of correct breaker alignments and "voltage" to required AC, DC, and 120 VAC instrument bus electrical power distribution subsystems. This changes the CTS by requiring the verification of the correct voltages and breaker alignments to the required AC, DC, and 120 VAC instrument bus electrical power distribution subsystems, whereas the CTS only requires verification of indicated power availability.

The purpose of this Surveillance is to ensure required electrical buses are properly energized to support their required function. This is more appropriately accomplished by verifying the correct breaker alignment and proper voltage is supplied to the required AC, DC, and 120 VAC instrument bus electrical power distribution subsystems. This change is acceptable because the Surveillance will continue to verify OPERABILITY of the required AC, DC and instrument bus electrical power distribution subsystems. Proper voltage from the required subsystems ensures proper voltage is supplied to the required safety features. This change is designated as more restrictive because the ITS requires verification of the correct breaker alignments and voltage, whereas the CTS only requires a verification of indicated power availability.

M03 **Unit 1 only:** CTS 3.8.2.3 Action a requires the unit to be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours when one of the required battery banks or buses is inoperable and cannot be restored to OPERABLE status within 2 hours. There is also an additional statement that LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN. ITS 3.8.9 ACTION D requires the unit to be in MODE 3 within 6 hours and MODE 5 within 36 hours. This changes the CTS by requiring the unit to be placed in COLD SHUTDOWN within the following 30 hours instead of HOT SHUTDOWN within the following 6 hours and eliminates the note associated with LCO 3.0.4.a.

The purpose of the CTS Actions is to ensure when an inoperable battery bank or bus cannot be restored to OPERABLE status, that the unit is placed in a MODE in which overall plant risk is minimized. PSL adopted TSTF-422, "Change in Technical Specifications End States (CE NPSD-1186)," Revision 2, with application of site-specific variations and deviations from TSTF-422, on August 30, 2016, in License Amendment 234 (NRC ADAMS Accession No. ML16210A374). The CTS Action was revised in Amendment 234 applying the end state action to one required battery bank or DC bus. However, TSTF-422 and the associated Combustion Engineering Topical Report CE NPSD-1186-A, “Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs,” Revision 00 (NRC ADAMS Accession No. ML110410539), did not include an evaluation of applying the end state to an inoperable DC bus. Therefore, when one DC electrical power distribution subsystem (i.e., required DC bus) is inoperable and cannot be restored within 2 hours, ITS 3.8.9 ACTION D requires the unit to be in MODE 3 (Hot Standby) within 6 hours and MODE 5 (Cold Shutdown) within 36 hours. As a result of the end state change, the Note associated with LCO 3.0.4.a is also unnecessary and deleted. This change is consistent with the ISTS and

TSTF-422. This change does not impact the allowance for an inoperable battery bank (Refer to ITS 3.8.4 for DC sources). The Completion Times to be in MODES 3 and 5 are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. This change is designated as more restrictive because the plant must be brought to a lower MODE than required by CTS when an inoperable DC electrical power distribution subsystem cannot be restored within the required Completion Time.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.8.2.1 and 3.8.2.3 require AC, AC instrument, and DC electrical buses to be OPERABLE and lists the specific busses, including applicable nominal AC bus voltages. Unit 2 CTS 3.8.3.1 requires specified electrical busses to be “energized in the specified manner” and lists the specified AC, DC, and 120 V instrument busses, including applicable nominal bus voltage.

ITS LCO 3.8.9 requires the applicable electrical power distribution subsystems; Train A and Train B AC, DC, and AC instrument buses to be OPERABLE. This changes the CTS by moving the specific names of the buses and the associated nominal bus voltages (i.e., 4160 V, 480 V, 120 V and 125 VDC) from the CTS to the ITS Bases.

The removal of these details related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.8.9 retains the requirement for the required electrical buses to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS 3.8.2.1 requires the AC electrical busses to be OPERABLE "and energized from sources of power other than the diesel generator sets.” Unit 1 CTS 4.8.2.1 also requires the AC buses to be determined OPERABLE "and energized from AC sources of power other than the diesel generator sets” by verifying indicated power availability. Unit 1 CTS 4.8.2.3.1 requires DC bus trains to be determined OPERABLE "and energized" by verifying indicated power availability. See DOC L01 for discussion of changes related to indicated power availability.

Unit 2 CTS LCO 3.8.3.1 requires the AC and DC electrical busses to "to be energized in the specified manner with both tie breakers open between redundant busses and between St. Lucie Unit 1 and Unit 2." Unit 2 CTS 4.8.3.1 also requires the specified buses to be determined "energized in the required manner" by verifying correct breaker alignment and indicated voltage.

ITS LCO 3.8.9 requires the applicable electrical power distribution subsystems to be OPERABLE and ITS SR 3.8.9.1 requires the verification of correct breaker alignment and voltage to required AC, DC, and AC instrument bus electrical power distribution subsystems. This changes the Unit 1 CTS by moving the procedural detail that the AC buses must be "energized from AC sources of power other than the diesel generator sets" and that the DC busses must be "energized" from the CTS to the ITS Bases. This changes the Unit 2 CTS by moving the procedural detail that the specified busses must be "energized in the specified manner with both tie breakers open between redundant busses and between St. Lucie Unit 1 and Unit 2."

The removal of these details for meeting Technical Specification requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the electrical power distribution subsystems to be OPERABLE and requires the verification of correct breaker alignment and voltage to required AC, DC, and AC instrument bus electrical power distribution subsystems. Also, this change is acceptable because these type of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* Unit 1 CTS 3.8.2.1 specifies that the AC electrical busses shall be OPERABLE "and energized from sources of power other than the diesel generator sets." Unit 2 CTS 3.8.3.1 requires electrical buses to be energized in a specified manner as listed in CTS 3.8.3.1.a through CTS 3.8.3.1.h. ITS LCO 3.8.9 does not include the requirement that an electrical bus be energized from a specific source of power. With regard to power sources, the ITS definition of OPERABLE includes the clarification that "normal or emergency electrical power sources" are acceptable. The CTS definition for OPERABLE simply states "electrical power" as a necessary component for OPERABILITY. This changes the CTS by removing the requirement that electrical busses be energized from specific sources of power.

The purpose of ITS 3.8.9 is to ensure required AC, DC, and AC instrument bus electrical power distribution subsystems are available to support the OPERABILITY of required systems, equipment and components and ensure the availability of sufficient power to operate the unit in a safe manner to mitigate the

consequences of postulated events. ITS 3.8.9 retains the requirement for the required electrical buses to be OPERABLE. The change is acceptable since it is consistent with the definition of OPERABILITY and the ability of the subsystem to support the dependent components and systems is not dependent of the specific source of the power provided the power source is a qualified electrical power source. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

3.8.2.1
3.8.2.3
DOC L01

LCO 3.8.9 Train A and Train B AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems shall be OPERABLE. (1)

instrument

Applicability

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystems inoperable.	A.1 ----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," for DC trains made inoperable by inoperable power distribution subsystems. ---------- Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ (2)
B. One or more AC ~~vital~~ buses inoperable.	B.1 Restore AC ~~vital~~ bus subsystem(s) to OPERABLE status.	2 hours (1) R1 [OR In accordance with the Risk Informed Completion Time Program~~]~~ (2)

ACTION NOTE

Action a

Action b

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a C. One ~~or more~~ DC electrical power distribution subsystem~~s~~ inoperable.	C.1 Restore DC electrical power distribution subsystem~~(s)~~ to OPERABLE status.	2 hours (1) ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ (2)
Action a Action b D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
DOC M01 E. Two or more electrical power distribution subsystems inoperable that result in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately

R1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 4.8.2.1 SR 4.8.2.3.1 DOC L01 SR 3.8.9.1	Verify correct breaker alignments and voltage to ~~[~~required~~]~~ AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems. (instrument)	~~[7 days~~ (1) ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

3.8.3.1 LCO 3.8.9 Train A and Train B AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems shall be OPERABLE. (1)

Applicability APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystems inoperable.	A.1 ----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," for DC trains made inoperable by inoperable power distribution subsystems. ---------- Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours [OR In accordance with the Risk Informed Completion Time Program] (2)
B. One or more AC ~~vital~~ buses inoperable.	B.1 Restore AC ~~vital~~ bus subsystem(s) to OPERABLE status.	2 hours [OR In accordance with the Risk Informed Completion Time Program] (1) (2) R1

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Action c C. One ~~or more~~ DC electrical power distribution subsystem~~s~~ inoperable.	C.1 Restore DC electrical power distribution subsystem~~(s)~~ to OPERABLE status.	2 hours (1) ~~[OR~~ ~~In accordance with the Risk Informed Completion Time Program]~~ (2)
Action a Action b Action c D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
DOC M01 E. Two or more electrical power distribution subsystems inoperable that result in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately

R1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 4.8.3.1 SR 3.8.9.1	Verify correct breaker alignments and voltage to ~~[~~required~~]~~ AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems. (instrument)	~~[7 days~~ (1) ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.9, DISTRIBUTION SYSTEMS - OPERATING

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems - Operating

BASES

BACKGROUND

The onsite Class 1E AC, DC, and AC ~~vital~~ bus electrical power distribution systems are divided by train into ~~[~~two~~]~~ redundant and independent AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems.

The AC electrical power subsystem for each train consists of a 4.16 kV ~~Engineered Safety Feature (ESF)~~ bus, having at least ~~[~~one separate and independent offsite source of power~~]~~ as well as a dedicated onsite diesel generator (DG) source. Each ~~[~~4.16 kV ~~ESF~~ bus~~]~~ is normally connected to ~~a preferred offsite source~~. After a loss of the ~~preferred offsite~~ power source to a 4.16 kV ~~ESF~~ bus, a transfer to the ~~alternate~~ offsite source is accomplished by utilizing ~~a time delayed bus undervoltage relay~~. If ~~all~~ offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ~~ESF~~ bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries. Additional description of this system may be found in the Bases for LCO 3.8.1, "AC Sources - Operating," and the Bases for LCO 3.8.4, "DC Sources - Operating."

The ~~secondary~~ AC electrical power distribution subsystem for each train includes the safety related ~~load centers, motor control centers, and distribution panels~~ shown in Table B 3.8.9-1.

The 120 VAC ~~vital~~ bus~~es~~ ~~are~~ arranged in two load groups per train and are normally powered from the inverters. The alternate power supply for the ~~vital~~ buses are Class 1E constant voltage source transformers powered from the same train as the associated inverter, and its use is governed by LCO 3.8.7, "Inverters - Operating." Each constant voltage source transformer is powered from a Class 1E AC bus.

The DC electrical power distribution subsystem consists of ~~[~~125~~]~~ V bus~~(~~es~~) and distribution panel(s)~~ (one for each train) with an interconnecting swing bus. The list of all required DC and ~~vital AC distribution~~ buses ~~[and panels]~~ is presented in Table B 3.8.9-1.

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 1) and Chapter ~~[~~15~~]~~ (Ref. 2), assume ~~ESF~~ systems are OPERABLE. The AC, DC, and AC ~~vital~~ bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC, DC, and AC ~~vital~~ bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC electrical power and

b. A worst case single failure.

The distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of AC, DC, and AC ~~vital~~ bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems are required to be OPERABLE.

Maintaining the Train A and Train B AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

OPERABLE AC electrical power distribution subsystems require the associated buses~~, load centers, motor control centers, and distribution panels~~ to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses ~~and distribution panels~~ to be energized to their proper voltage from either the associated battery or charger. OPERABLE ~~vital~~ bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated ~~[~~inverter via inverted DC voltage~~,~~ ~~inverter using internal AC source,~~ or Class 1E constant voltage transformer~~]~~.

BASES

LCO (continued)

In addition, tie breakers between redundant safety related AC, DC, and AC ~~vital~~ bus power distribution subsystems~~, if they exist,~~ must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. ~~It does not, however, preclude redundant Class 1E 4.16 kV buses from being powered from the same offsite circuit.~~

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.10, "Distribution Systems - Shutdown."

ACTIONS

A.1

With one or more Train A and B required AC ~~buses, load centers, motor control centers, or distribution panels~~ (except AC ~~vital~~ buses), in one train inoperable and a loss of function has not occurred, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses~~, load centers, motor control centers, and distribution panels~~ must be restored to OPERABLE status within 8 hours ~~[or in accordance with the Risk Informed Completion Time Program]~~.

1

Insert 1

Based on the number of safety significant electrical loads associated with each bus listed in Table B 3.8.9-1, if one or more of the buses becomes inoperable, entry into the appropriate ACTIONS of LCO 3.8.9 is required. Some buses, such as distribution panels and motor control centers, which help comprise the AC and DC distribution systems, are not listed in Table B 3.8.9-1. The loss of electrical loads associated with these buses may not result in a complete loss of a redundant safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should one or more of these buses become inoperable due to a failure not affecting the OPERABILITY of a bus listed in Table B 3.8.9-1 (e.g., a breaker supplying a single distribution panel fails open), the individual loads on the bus would be declared inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads would be entered. However, if one or more of these buses is inoperable due to a failure also affecting the OPERABILITY of a bus listed in Table B 3.8.9-1 (e.g., loss of 4.16 kV essential bus, which results in de-energization of all buses powered from the 4.16 kV essential bus), then the Conditions and Required Actions of LCO 3.8.9 must be entered. | R1

BASES

ACTIONS (continued)

Condition A worst scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train, to the actions associated with taking the unit to shutdown within this time limit and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

Required Action A.1 is modified by a Note that requires the applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," to be entered for DC trains made inoperable by inoperable power distribution subsystems. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. Inoperability of a distribution system can result in loss of charging power to batteries and eventual loss of DC power. This Note ensures that the appropriate attention is given to restoring charging power to batteries, if necessary, after loss of distribution systems.

B.1

instrument

instrument With one or more AC ~~vital~~ buses inoperable, and a loss of function has not yet occurred, the remaining OPERABLE AC ~~vital~~ buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the instrument ~~[~~required~~]~~ AC ~~vital~~ bus must be restored to OPERABLE status within 2 hours by powering the bus from the associated ~~[~~inverter via inverted DC~~, inverter using internal AC source,~~ or Class 1E constant voltage transformer~~]~~. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~

1

1 R1

2 R1

BASES

ACTIONS (continued)

Condition B represents one or more AC ~~vital~~ buses without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining ~~vital~~ buses, and restoring power to the affected ~~vital~~ bus.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate ~~vital~~ AC power. Taking exception to LCO 3.0.2 for components without adequate ~~vital~~ AC power, which would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue,

b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate ~~vital~~ AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train, and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the AC ~~vital~~ bus to OPERABLE status, the redundant capability afforded by the other OPERABLE ~~vital~~ buses, and the low probability of a DBA occurring during this period.

C.1

With one ~~or more~~ DC ~~buses or distribution panels~~ inoperable, and a loss of function has not yet occurred, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the ~~[required]~~ DC ~~buses and~~

BASES

ACTIONS (continued)

~~distribution panels~~ must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger. 1

~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 2

Condition C represents one ~~or more DC buses or distribution panels~~ without adequate DC power; potentially both with the battery significantly degraded and the ~~associated~~ charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train. 1

electrical power distribution subsystem

s

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue,

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train, and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for the electrical power distribution subsystem DC ~~buses~~ is consistent with Regulatory Guide 1.93 (Ref. 3). 1

the

electrical power distribution subsystem

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

E.1

Condition E corresponds to a level of degradation in the electrical distribution system that causes a required safety function to be lost. When more than one inoperable electrical power distribution subsystem results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

instrument

This Surveillance verifies that the AC, DC, and AC ~~vital~~ bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. ~~[The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

REFERENCES

1. FSAR, Chapter ~~[~~6~~]~~.

U

2. FSAR, Chapter ~~[~~15~~]~~.

3. Regulatory Guide 1.93, December 1974.

Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	TRAIN A*	TRAIN B*	
emergency AC ~~safety~~ buses	Emergency ~~[~~4160 V~~]~~	1A3 ~~[~~ESF Bus~~]~~ ~~[NB01]~~	Emergency ~~[~~ESF Bus~~]~~ ~~[NB02]~~ 1B3	1
	Emergency Bus ~~[~~480 V~~]~~	1A2 ~~Load Centers [NG01, NG03]~~	1B2 ~~Load Centers [NG02, NG04]~~	1
	~~[~~480 V~~]~~ 1A5, 1A6, 1A7	MCC Buses ~~Motor Control Centers [NG01A, NG01I, NG01B, NG03C, NG03I, NG03D]~~	MCC Buses ~~Motor Control Centers [NG02A, NG02I, NG02B, NG04C, NG04I, NG04D]~~ 1B5, 1B6, 1B7	1
	~~[120 V]~~	~~Distribution Panels [NP01, NP03]~~	~~Distribution Panels [NP02, NP04]~~	2
DC buses	~~[~~125 V~~]~~	1A Bus ~~[NK01]~~ ~~Bus [NK03]~~ ~~Distribution Panels [NK41, NK43, NK51]~~	1B Bus ~~[NK02]~~ ~~Bus [NK04]~~ ~~Distribution Panels [NK42, NK44, NK52]~~	
instrument AC ~~vital~~ buses	~~[~~120 V~~]~~	1MA Bus ~~[NN01]~~ 1MC Bus ~~[NN03]~~	1MB Bus ~~[NN02]~~ 1MD Bus ~~[NN04]~~	1

* Each train of the AC and DC electrical power distribution systems is a subsystem.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems - Operating

BASES

BACKGROUND

The onsite Class 1E AC, DC, and AC ~~vital~~ bus electrical power distribution systems are divided by train into ~~[~~two~~]~~ redundant and independent AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems.

The AC electrical power subsystem for each train consists of a 4.16 kV ~~Engineered Safety Feature (ESF)~~ bus, having at least ~~[~~one separate and independent offsite source of power~~]~~ as well as a dedicated onsite diesel generator (DG) source. Each ~~[~~4.16 kV ~~ESF~~ bus~~]~~ is normally connected to ~~a preferred offsite source~~. After a loss of the ~~preferred offsite~~ power source to a 4.16 kV ~~ESF~~ bus, a transfer to the ~~alternate~~ offsite source is accomplished by utilizing ~~a time delayed bus undervoltage relay~~. If ~~all~~ offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ~~ESF~~ bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries. Additional description of this system may be found in the Bases for LCO 3.8.1, "AC Sources - Operating," and the Bases for LCO 3.8.4, "DC Sources - Operating."

The ~~secondary~~ AC electrical power distribution subsystem for each train includes the safety related ~~load centers, motor control centers, and distribution panels~~ shown in Table B 3.8.9-1.

The 120 VAC ~~vital~~ bus~~es~~ ~~are~~ arranged in two load groups per train and are normally powered from the inverters. The alternate power supply for the ~~vital~~ buses are Class 1E constant voltage source transformers powered from the same train as the associated inverter, and its use is governed by LCO 3.8.7, "Inverters - Operating." Each constant voltage source transformer is powered from a Class 1E AC bus.

The DC electrical power distribution subsystem consists of ~~[~~125~~]~~ V bus~~(es) and distribution panel(s)~~.

The list of all required DC and ~~vital AC distribution~~ buses ~~[and panels]~~ is presented in Table B 3.8.9-1.

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 1) and Chapter ~~[~~15~~]~~ (Ref. 2), assume ~~ESF~~ systems are OPERABLE. The AC, DC, and AC ~~vital~~ bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC, DC, and AC ~~vital~~ bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC electrical power and

b. A worst case single failure.

The distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of AC, DC, and AC ~~vital~~ bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems are required to be OPERABLE.

Maintaining the Train A and Train B AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

OPERABLE AC electrical power distribution subsystems require the associated buses~~, load centers, motor control centers, and distribution panels~~ to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses ~~and distribution panels~~ to be energized to their proper voltage from either the associated battery or charger. OPERABLE ~~vital~~ bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated ~~[~~inverter via inverted DC voltage~~,~~ ~~inverter using internal AC source,~~ or Class 1E constant voltage transformer~~]~~.

BASES

LCO (continued)

R1

instrument

In addition, tie breakers between redundant safety related AC, DC, and AC ~~vital~~ bus power distribution subsystems~~, if they exist,~~ must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. ~~It does not, however, preclude redundant Class 1E 4.16 kV buses from being powered from the same offsite circuit.~~

1

1

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.10, "Distribution Systems - Shutdown."

ACTIONS

A.1

electrical power distribution subsystems

instrument

With one or more Train A and B required AC ~~buses, load centers, motor control centers, or distribution panels~~ (except AC ~~vital~~ buses), in one train inoperable and a loss of function has not occurred, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses~~, load centers, motor control centers, and distribution panels~~ must be restored to OPERABLE status within 8 hours ~~[or in accordance with the Risk Informed Completion Time Program]~~.

1

2

1

1

~~Insert 1~~

~~Based on the number of safety significant electrical loads associated with each bus listed in Table B 3.8.9-1, if one or more of the buses becomes inoperable, entry into the appropriate ACTIONS of LCO 3.8.9 is required. Some buses, such as distribution panels and motor control centers, which help comprise the AC and DC distribution systems, are not listed in Table B 3.8.9-1. The loss of electrical loads associated with these buses may not result in a complete loss of a redundant safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should one or more of these buses become inoperable due to a failure not affecting the OPERABILITY of a bus listed in Table B 3.8.9-1 (e.g., a breaker supplying a single distribution panel fails open), the individual loads on the bus would be declared inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads would be entered. However, if one or more of these buses is inoperable due to a failure also affecting the OPERABILITY of a bus listed in Table B 3.8.9-1 (e.g., loss of 4.16 kV essential bus, which results in de-energization of all buses powered from the 4.16 kV essential bus), then the Conditions and Required Actions of LCO 3.8.9 must be entered.~~ |R1

BASES

ACTIONS (continued)

Condition A worst scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train, to the actions associated with taking the unit to shutdown within this time limit and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

Required Action A.1 is modified by a Note that requires the applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," to be entered for DC trains made inoperable by inoperable power distribution subsystems. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. Inoperability of a distribution system can result in loss of charging power to batteries and eventual loss of DC power. This Note ensures that the appropriate attention is given to restoring charging power to batteries, if necessary, after loss of distribution systems.

B.1

instrument

instrument

With one or more AC ~~vital~~ buses inoperable, and a loss of function has not yet occurred, the remaining OPERABLE AC ~~vital~~ buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the ~~[~~required~~]~~ AC ~~vital~~ bus must be restored to OPERABLE status within 2 hours by powering the bus from the associated ~~[~~inverter via inverted DC~~, inverter using internal AC source,~~ or Class 1E constant voltage transformer~~]~~. ~~[~~Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.~~]~~

instrument

1

1

2

R1

BASES

ACTIONS (continued)

Condition B represents one or more AC ~~vital~~ buses without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining ~~vital~~ buses, and restoring power to the affected ~~vital~~ bus.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate ~~vital~~ AC power. Taking exception to LCO 3.0.2 for components without adequate ~~vital~~ AC power, which would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue,

b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate ~~vital~~ AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train, and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the AC ~~vital~~ bus to OPERABLE status, the redundant capability afforded by the other OPERABLE ~~vital~~ buses, and the low probability of a DBA occurring during this period.

C.1

With one ~~or more~~ DC ~~buses or distribution panels~~ inoperable, and a loss of function has not yet occurred, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the ~~[required]~~ DC ~~buses and~~

BASES

ACTIONS (continued)

~~distribution panels~~ must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger. 1

~~[Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.]~~ 2

Condition C represents one or more DC ~~buses or distribution panels~~ without adequate DC power; potentially both with the battery significantly degraded and the ~~associated~~ charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train. 1

electrical power distribution subsystem

s

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue,

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train, and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for the electrical power distribution subsystem DC ~~buses~~ is consistent with Regulatory Guide 1.93 (Ref. 3). 1

the

electrical power distribution subsystem

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

E.1

Condition E corresponds to a level of degradation in the electrical distribution system that causes a required safety function to be lost. When more than one inoperable electrical power distribution subsystem results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the AC, DC, and AC ~~vital~~ instrument bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. ~~[The 7-day Frequency takes into account the redundant capability of the AC, DC, and AC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

REFERENCES

1. FSAR, Chapter ~~[~~6~~]~~.
2. FSAR, Chapter ~~[~~15~~]~~.
3. Regulatory Guide 1.93, December 1974.

Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	TRAIN A*	TRAIN B*
emergency AC ~~safety~~ buses	Emergency ~~[~~4160 V~~]~~	2A3 ~~[ESF~~ Bus~~]~~ ~~[NB01]~~	Emergency ~~[ESF~~ Bus~~]~~ ~~[NB02]~~ 2B3
	Emergency Bus ~~[~~480 V~~]~~	2A2, 2A5 ~~Load Centers [NG01, NG03]~~	2B2, 2B5 ~~Load Centers [NG02, NG04]~~
	~~[~~480 V~~]~~ 2A5, 2A6, 2A7, 2A8, 2A9	MCC Buses ~~Motor Control Centers [NG01A, NG01I, NG01B, NG03C, NG03I, NG03D]~~	MCC Buses ~~Motor Control Centers [NG02A, NG02I, NG02B, NG04C, NG04I, NG04D]~~ 2B5, 2B6, 2B7, 2B8, 2B9
	~~[120 V]~~	~~Distribution Panels [NP01, NP03]~~	~~Distribution Panels [NP02, NP04]~~
DC buses	~~[~~125 V~~]~~	2A Bus ~~[NK01]~~ ~~Bus [NK03]~~ ~~Distribution Panels [NK41, NK43, NK51]~~	2B Bus ~~[NK02]~~ ~~Bus [NK04]~~ ~~Distribution Panels [NK42, NK44, NK52]~~
instrument AC ~~vital~~ buses	~~[~~120 V~~]~~	2MA Bus ~~[NN01]~~ 2MC Bus ~~[NN03]~~	2MB Bus ~~[NN02]~~ 2MD Bus ~~[NN04]~~

1 1 1 2 1

* Each train of the AC and DC electrical power distribution systems is a subsystem.

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.9 BASES, DISTRIBUTION SYSTEMS - OPERATING

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.8.9, DISTRIBUTION SYSTEMS - OPERATING

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 10

ITS 3.8.10, Distribution Systems - Shutdown

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

The necessary portion of
and AC instrument bus power distribution subsystems
to support equipment required to be OPERABLE.

M01

LA02

L01

LCO 3.8.10 3.8.2.2 ~~As a minimum, the following~~ A.C. ~~.~~ electrical ~~busses~~ shall be OPERABLE ~~and energized from sources of power other than a diesel generator set but aligned to an OPERABLE diesel generator set:~~

~~1 - 4160 volt Emergency Bus~~

~~1 - 480 volt Emergency Bus~~

~~3 - 480 volt Emergency MCC Busses~~

~~2 - 120 volt A.C. Instrument Busses~~

LA01

Applicability **APPLICABILITY**: MODES 5 and 6 ~~,~~

During movement of irradiated fuel assemblies.

M02

ACTION:

Add proposed ACTIONS NOTE

With less than the above complement of A.C. busses OPERABLE ~~and energized, establish CONTAINMENT INTEGRITY within 8 hours~~.

LA02

M04 L02

Add proposed Required Actions A.2.1, A.2.2, A.2.3 and A.2.4

Add proposed Required Action A.1

M01

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1 4.8.2.2 ~~The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators~~ in accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing indicated power availability~~.

LA02

L01

M03

correct breaker alignments and voltage to required AC, DC, and AC instrument bus electrical distribution subsystems

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

LCO 3.8.10

3.8.2.4 ~~As a minimum, the following~~ D~~.~~C~~.~~ electrical equipment ~~and bus~~ shall be ~~energized and~~ OPERABLE:

~~1 - 125-volt D.C. bus, and~~

1 - 125-volt battery bank and charger supplying the above D.C. bus.

Applicability

APPLICABILITY: MODES 5 and 6~~:~~ , During movement of irradiated fuel assemblies.

ACTION:

Add proposed ACTIONS NOTE

With less than the above complement of D~~.~~C~~.~~ equipment ~~and~~ bus OPERABLE, ~~establish CONTAINMENT INTEGRITY within 8 hours~~.

Add proposed Required Actions A.2.1, A.2.2, A.2.3 and A.2.4

Add proposed Required Action A.1

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

4.8.2.4.1 ~~The above required 125-volt D.C. bus shall be determined OPERABLE and energized~~ in accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing indicated power availability~~.

4.8.2.4.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

The necessary portion of

power distribution subsystems

See ITS 3.8.5

See ITS 3.8.5

See ITS 3.8.5

See ITS 3.8.6

correct breaker alignments and voltage to required AC, DC, and AC instrument bus electrical distribution subsystems

M01

LA02

LA01

M02

M04

L02

M01

LA02

M03

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

The necessary portion of AC, DC, and AC instrument bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

M01 L01 LA02

LCO 3.8.10 3.8.3.2 ~~As a minimum, the following electrical busses shall be energized and in the specified manner:~~

~~a. One train of A.C. emergency busses consisting of one 4160 volt and two 480 volt A.C. emergency busses.~~

~~b. Two 120 volt A.C. Instrument Busses energized from their associated inverters connected to their respective D.C. busses.~~

~~c. One 125 volt D.C. bus energized from its associated battery bank.~~

LA01 L01 L01

Applicability **APPLICABILITY**: MODES 5 and 6~~.~~ ,

During movement of irradiated fuel assemblies.

M02

ACTION:

ACTION A Add proposed ACTIONS NOTE

Add proposed Required Action A.1

inoperable

M01 L01 L03

With any of the above ~~required electrical busses not energized in the required manner,~~ immediately suspend ~~all operations involving CORE ALTERATIONS,~~ operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, or movement of irradiated fuel, initiate ~~corrective~~ action to ~~energize the required electrical busses in the specified manner as soon as possible, and within 8 hours depressurize and vent the RCS through a 3.58 square inch vent.~~

ACTION A.2.2

ACTION A.2.1

ACTION A.2.3

restore required AC, DC, and AC instrument bus electrical power distribution subsystems to OPERABLE status.

L04 M04

Add proposed Required Action A.2.4

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1 4.8.3.2 ~~The specified busses shall be determined energized in the required manner~~ in accordance with the Surveillance Frequency Control Program ~~by~~ verify~~ing~~ correct breaker alignment and ~~indicated~~ voltage ~~on the busses~~.

L01 LA02

to required AC, DC, and AC instrument bus electrical distribution subsystems

~~DELETED~~

~~DELETED~~

~~DELETED~~

DISCUSSION OF CHANGES
ITS 3.8.10, DISTRIBUTION SYSTEMS - SHUTDOWN

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications – Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 Unit 1 CTS 3.8.2.2 lists the minimum AC and AC instrument buses required to be OPERABLE. Unit 1 CTS 3.8.2.4 requires one 125 VDC bus to be OPERABLE. Unit 2 CTS 3.8.3.2 lists the minimum AC, DC, and AC instrument buses required to be OPERABLE and energized in the manner specified therein.

ITS LCO 3.8.10 requires the necessary portion of AC, DC, and AC instrument bus electrical power subsystems must be OPERABLE to support equipment required to be OPERABLE. In addition, an optional Required Action (ITS 3.8.10 Required Action A.1) has been added which allows the associated supported required feature(s) to be declared inoperable. This changes CTS by adding a requirement that the applicable portions of the AC, DC, and AC instrument bus electrical power distribution subsystems must be OPERABLE when required to support equipment required to be OPERABLE by the Technical Specifications. This could require more buses to be OPERABLE than is currently required based on plant conditions. In addition, an action has been added to allow an option to the existing actions to compensate for the additional required electrical buses.

The purpose of Unit 1 CTS 3.8.2.2 and 3.8.2.4 and Unit 2 CTS 3.8.3.2 is to ensure that the minimum necessary A.C. and D.C. distribution systems are OPERABLE to support shutdown and refueling for extended periods with sufficient instrumentation and controls available for monitoring and maintaining plant status. This change adds a requirement that the necessary portions of AC, DC, and AC instrument bus electrical power distribution subsystems must be OPERABLE when required to support the equipment required to be OPERABLE by the Technical Specifications. This added restriction conservatively assures the needed electrical power distribution buses are OPERABLE, even if it results in more electrical power distribution systems being required. Since the ITS 3.8.10 electrical power distribution subsystem OPERABILITY requirements require the necessary portions of the distribution subsystems to be OPERABLE to support equipment required to be OPERABLE, if a portion of the electrical power distribution subsystem cannot supply any required equipment, that electrical power distribution subsystem is inoperable. In this event it may not be necessary to suspend irradiated fuel handling and positive reactivity additions. Conservative actions can be assured if the required equipment without the necessary power is declared inoperable, and the associated ACTIONS of the individual equipment is taken (ITS 3.8.10 Required Action A.1). Therefore, along with the conservative

additional requirements placed on the electrical power distribution subsystems, Required Action A.1, requires the associated supported equipment to be declared inoperable. These changes are acceptable since the additions represent restrictions consistent with implicit assumptions for operation in shutdown conditions (required equipment receiving the necessary required power), and these restrictions are not currently imposed by the Technical Specifications. This change is designated as more restrictive because it adds a new requirement to the CTS.

M02 Unit 1 CTS 3.8.2.2 and CTS 3.8.2.4 and Unit 2 CTS 3.8.3.2 are applicable in MODES 5 and 6. ITS 3.8.10 is applicable in MODE 5 and 6 and during movement of irradiated fuel assemblies. In addition, a Note has been added to the ACTIONS which states that LCO 3.0.3 is not applicable. This changes the CTS by adding the Applicability of “during movement of irradiated fuel assemblies” and adds an ACTIONS Note stating that LCO 3.0.3 is not applicable. | R1

The purpose of CTS Applicability is to ensure that sufficient electrical power subsystems are OPERABLE to support equipment required to mitigate the consequences of an analyzed event during shutdown modes. This change is acceptable because the proposed requirements are necessary to ensure the electrical power subsystems are OPERABLE to support equipment required to OPERABLE during movement of irradiated fuel assemblies. Movement of fuel normally occurs during MODES 5 and 6, however, it can also occur outside of containment in other plant MODES (MODES 1, 2, 3, and 4) or other conditions (i.e., reactor defueled). This Specification is needed to ensure the appropriate distribution system requirements are specified during fuel handling and ensure the appropriate actions are taken (i.e., stop fuel movement) when the minimum electrical supply is not available (See Unit 1 DOC L02 and Unit 2 DOCS L03 and L04 for additional changes to the Required Actions). This change adds a clarification Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODES 5 or 6, LCO 3.0.3 is not applicable and would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations and the inability to suspend movement in accordance with the ITS 3.8.10 Required Actions would not be sufficient reason to require a reactor shutdown. The Note has been added for clarification and is necessary since defaulting to LCO 3.0.3 would require the reactor to be shutdown but would not require suspension of the activities with a potential for releasing radioactive materials. This change is designated as more restrictive because the ITS requires the equipment to be OPERABLE during movement of irradiated fuel assemblies both inside and outside of the containment in conditions other than MODES 5 and 6.

M03 **Unit 1 only:** CTS 4.8.2.2 requires, in part, the specified A.C. busses shall be determined OPERABLE by verifying "indicated power availability" and CTS 4.8.2.4.1 states each D.C. bus train shall be determined OPERABLE by verifying "indicated power availability." ITS SR 3.8.10.1 requires the verification of correct breaker alignments and "voltage" to required AC, DC, and 120 VAC instrument bus electrical power distribution subsystems. This changes the CTS by requiring the verification of the correct voltages and breaker alignments to the required AC, DC, and 120 VAC instrument bus electrical power distribution subsystems, whereas the CTS only requires verification of indicated power availability.

The purpose of this Surveillance is to ensure required electrical buses are properly energized to support their required function. This is more appropriately accomplished by verifying the correct breaker alignment and proper voltage is supplied to the required AC, DC, and 120 VAC instrument bus electrical power distribution subsystems. This change is acceptable because the Surveillance will continue to verify OPERABILITY of the required AC, DC and instrument bus electrical power distribution subsystems. Proper voltage from the required subsystems ensures proper voltage is supplied to the required safety features. This change is designated as more restrictive because the ITS requires verification of the correct breaker alignments and voltage, whereas the CTS only requires a verification of indicated power availability.

M04 With less than the minimum specified AC and DC busses energized, Unit 2 CTS 3.8.3.2 action requires, in part, the immediate suspension of operations involving positive reactivity additions that could result in loss of required SDM or boron concentration or movement of irradiated fuel. Unit 1 CTS 3.8.2.2 and 3.8.2.4 require establishing CONTAINMENT INTEGRITY within 8 hours. ITS 3.8.10 includes an additional required action (Required Action A.2.4) to "declare associated required shutdown cooling subsystem(s) inoperable and not in operation." This changes the CTS by adding a requirement to declare the required shutdown cooling subsystem(s) (SDCs) inoperable and not in operation that is not currently specifically imposed by the Technical Specification.

This change is necessary because the CTS actions do not adequately address the concern relating to coolant circulation and heat removal. Pursuant to CTS 3.0.6 (ITS LCO 3.0.6), the SDC actions would not be entered unless a loss of safety function was determined to exist. Therefore, Required Action A.2.4 is added to immediately take appropriate SDC actions. This change is designated as more restrictive because it adds a new requirement to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.8.2.2 and 3.8.2.4 require AC AC instrument, and DC electrical buses to be OPERABLE and lists the specific busses, including applicable nominal AC bus voltages. Unit 2 CTS 3.8.3.2 requires electrical busses to be "energized in the specified manner" and lists the specified AC, DC, and 120 V instrument busses, including applicable nominal bus voltage.

ITS LCO 3.8.10 requires the necessary portion of AC, DC, and AC instrument bus electrical power subsystems must be OPERABLE to support equipment required to be OPERABLE. This changes the CTS by moving the specific names of the buses and the associated nominal bus voltages (i.e., 4160 V, 480 V, 120 V and 125 VDC) from the CTS to the ITS Bases.

The removal of these details related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.8.10 retains the requirement for the required electrical buses to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS 3.8.2.2 requires specific AC electrical busses to be OPERABLE "and energized from sources of power other than a diesel generator set but aligned to an OPERABLE diesel generator set.” Unit 1 CTS 4.8.2.2 also requires the specified AC buses to be determined OPERABLE "and energized from AC sources of power other than the diesel generators” by verifying indicated power availability. Unit 1 CTS 3.8.2.4 requires the DC bus to be “energized” and OPERABLE. Unit 1 CTS 4.8.2.4.1 requires DC busses to be determined OPERABLE and "energized" by verifying indicated power availability.

Unit 2 CTS LCO 3.8.3.2 requires AC and DC electrical busses to "to be energized in the specified manner.” Unit 2 CTS 4.8.3.2 requires the specified buses to be determined "energized in the required manner” by verifying correct breaker alignment and indicated voltage.

ITS LCO 3.8.10 requires the necessary portions of the electrical power distribution subsystems to be OPERABLE to support equipment required to be OPERABLE and ITS SR 3.8.10.1 requires the verification of correct breaker alignments and voltage to required AC, DC, and AC instrument bus electrical power distribution subsystems.

This changes the Unit 1 and Unit 2 CTS by moving the procedural detail that the AC buses and DC busses must be “energized” from the CTS to the ITS Bases. See DOC L01 for discussion of additional changes to the respective CTS LCO and Surveillance Requirement.

The removal of these details for meeting Technical Specification requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for the electrical power distribution subsystems to be OPERABLE and requires the verification of correct breaker alignments and voltage to required AC, DC, and AC instrument bus electrical power distribution subsystems. Also, this change is acceptable because these type of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail

change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* Unit 1 CTS 3.8.2.2 specifies that the AC electrical busses shall be OPERABLE "and energized from sources of power other than the diesel generator sets but aligned to an OPERABLE diesel generator set." Unit 2 CTS 3.8.3.2 requires that specified electrical busses be energized "and in the specified manner," as listed in CTS 3.8.3.2.a through CTS 3.8.3.2.c. ITS LCO 3.8.10 does not include the requirement that the busses be energized by any specific method. ITS defines OPERABLE to include the clarification that "normal or emergency electrical power sources" are acceptable. The CTS definition for OPERABLE simply states "electrical power" as a necessary component for OPERABILITY. This changes the CTS by removing the requirement that electrical busses be energized in any specific manner or from a specific source of power.

The purpose of ITS 3.8.10 is to ensure the necessary AC, DC, and AC instrument bus electrical power distribution subsystems are available to support the OPERABILITY of required systems, equipment and components and ensure the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown. ITS 3.8.10 retains the requirement for the required electrical buses to be OPERABLE. The change is acceptable since it is consistent with the definition of OPERABILITY and the ability of the subsystem to support the dependent components and systems is independent of the specific source of the power provided the power source is a qualified electrical power source. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than applied in the CTS.

L02 **Unit 1 only:** *(Category 4 – Relaxation of Required Action)* With less than the minimum complement of electrical busses OPERABLE and energized, CTS 3.8.2.2 and 3.8.2.4 require the establishment of containment integrity within 8 hours. ITS 3.8.10 ACTION A requires suspending movement of irradiated fuel assemblies, suspension of operations involving positive reactivity additions that could result in the loss of required SDM or boron concentration, the initiation of actions to restore required AC, DC, and AC instrument bus electrical power distribution subsystems to OPERABLE status, and the declaration of the associated required shutdown cooling subsystems(s) inoperable and not in operation. This changes the CTS by replacing the existing Required Action to restore containment integrity with Actions that will prevent a fuel handling accident from occurring, ensure the reactor remains subcritical, and ensure decay heat removal is reestablished.

The purpose of the CTS 3.8.2.2 and 3.8.2.4 Actions is to isolate the containment to minimize any release from the plant if an event were to occur during shutdown conditions. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe

operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. The proposed Required Actions require the suspension of movement of irradiated fuel assemblies, the suspension of operations involving positive reactivity additions that could result in the loss of required SDM or boron concentration, the initiation of actions to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status, and the declaration of the associated required decay heat removal subsystems(s) inoperable and not in operation. Suspending the movement of irradiated fuel assemblies will prevent a fuel handling accident from occurring. Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. The requirement to initiate action to restore required AC, DC, and AC instrument bus electrical power distribution subsystems to OPERABLE status will place the plant in compliance with the LCO in an expeditious manner. Declaration of the associated required shutdown cooling subsystem(s) inoperable and not in operation will require the plant to enter the applicable LCOs to apply additional Required Actions. The proposed actions will also minimize the potential for any accident releases outside of the containment and are considered acceptable in lieu of the current action to restore containment integrity within 8 hours. The actions may be considered somewhat more restrictive since immediate action is required, however, is classified as less restrictive since the current actions to restore containment integrity have been deleted. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* CTS 3.8.3.2 ACTION states, in part, “With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity changes that could result in loss of required SHUTDOWN MARGIN or boron concentration, or movement of irradiated fuel.” ITS 3.8.10 ACTIONS do not include the requirement to suspend CORE ALTERATIONS. This changes the CTS by removing the Required Action to suspend all operations involving CORE ALTERATIONS.

The purpose of the CTS 3.8.3.2 ACTION is to minimize the possibility of an event that may require the electrical power source to mitigate the consequences of a design basis event. CORE ALTERATIONS is defined in CTS 1.9, in part, as "the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity, within the reactor vessel with the head removed and fuel in the vessel." Core Alterations only occur when the reactor vessel head is removed - it only applies in MODE 6. There is only one accident considered during MODE 6 that involves a Core Alteration - a fuel handling accident. According to UFSAR Section 15.7.4.1.2, a fuel handling accident is initiated by the dropping of an irradiated fuel assembly either in the containment or in the fuel handling building. Suspension of CORE ALTERATIONS, except for suspension of movement of irradiated fuel, will not prevent or impair the mitigation of a fuel handling accident. ITS 3.8.10 retains the requirement to suspend movement of irradiated fuel assemblies as Required Action A.2.1 (for one or more electrical distribution subsystems). R1

Deleting the reference to “CORE ALTERATIONS” is acceptable because the only CORE ALTERATION analyzed in the safety analysis that is potentially affected by an inoperable electrical distribution subsystem is covered by the ITS Required Actions. Additionally, CORE ALTERATIONS that may be capable of damaging a fuel assembly prior to the 72 hour decay period is improbable because of the physical time required to perform plant shutdown, cooldown, depressurize the Reactor Coolant System (RCS), and the additional operations required to expose irradiated fuel in the reactor vessel (e.g., containment entry, removal of vessel head, removal of vessel internals, etc.) is greater than 72 hours. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS. R1

L04 **Unit 2 only:** *(Category 4 – Relaxation of Required Action)* Unit 2 CTS 3.8.3.2 ACTION requires, in part, with any of the specified electrical busses not energized in the required manner, venting of the RCS through a 3.58 square inch vent within 8 hours. ITS 3.8.10 does not include the venting requirement.

The purpose of CTS is to ensure the minimum electrical power distribution buses are OPERABLE. CTS 3.4.9.3 provides specific OPERABILITY requirements for overpressure protection systems when in MODE 5 and MODE 6 with the reactor vessel head on. The definition of OPERABLE in CTS 1.19 requires necessary electrical power to components so they can perform their required safety function. This change is acceptable because CTS 3.4.9.3 Action a.1 requires depressurization and venting of the RCS when adequate pressure relief methods are inoperable, including the loss of electrical power necessary to support the required pressure relief components. ITS 3.4.12 retains the requirement to depressurize and establish the required RCS vent size (Refer to ITS 3.4.12 for any changes to CTS 3.4.9.3 requirements). Therefore, the CTS 3.8.1.2 action to depressurize the RCS and vent through a greater than or equal to 3.58 square inch vent is unnecessary. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

3.8.2.2
3.8.2.4
DOC M01
DOC L01

LCO 3.8.10 The necessary portion of AC, DC, and AC ~~vital~~ instrument bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE. (1)

Applicability
DOC M02

APPLICABILITY: MODES 5 and 6,
During movement of ~~[recently]~~ irradiated fuel assemblies. (2)

ACTIONS

----------NOTE----------

DOC M02 LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION DOC M01 DOC L02 A. One or more required AC, DC, or AC ~~vital~~ instrument bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable. (1) OR	Immediately
DOC L02	A.2.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies. (2) AND	Immediately
DOC L02	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. AND	Immediately

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L02	A.2.3 Initiate actions to restore required AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems to OPERABLE status. (instrument) (1)	Immediately
	AND	
DOC L02 DOC M04	A.2.4 Declare associated required shutdown cooling subsystem(s) inoperable and not in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.2.2 4.8.2.4.2 DOC L01 DOC M03 SR 3.8.10.1	Verify correct breaker alignments and voltage to required AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems. (instrument)	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (1) (2)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

3.8.3.2 DOC M01 DOC L01

LCO 3.8.10 The necessary portion of AC, DC, and AC ~~vital~~ instrument bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE. (1)

Applicability DOC M02

APPLICABILITY: MODES 5 and 6,
During movement of ~~[recently]~~ irradiated fuel assemblies. (2)

ACTIONS

----------NOTE----------

DOC M02 LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
ACTION DOC M02 DOC L03 A. One or more required AC, DC, or AC ~~vital~~ bus electrical power instrument distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable. (1) OR A.2.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies. (2) AND A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. AND	Immediately Immediately Immediately

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2.3 Initiate actions to restore required AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems to OPERABLE status. (instrument) (1)	Immediately
	AND	
DOC M04	A.2.4 Declare associated required shutdown cooling subsystem(s) inoperable and not in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.8.3.2 DOC L01 SR 3.8.10.1	Verify correct breaker alignments and voltage to required AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems. (instrument)	~~[7 days~~ (1) ~~OR~~ In accordance with the Surveillance Frequency Control Program~~]~~ (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.10, DISTRIBUTION SYSTEMS - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant-specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Bases Justification for Deviations (JFDs)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the AC, DC, and AC ~~vital~~ bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 1) and Chapter ~~[~~15~~]~~ (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and AC ~~vital~~ bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC ~~vital~~ bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems during MODES 5 and 6, and during movement of ~~[recently]~~ irradiated fuel assemblies, ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident ~~[involving handling recently irradiated fuel. Due to radioactive decay, AC, DC, and AC vital bus electrical power is only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~.

The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific unit condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution ~~system~~ necessary to support OPERABILITY of required systems, equipment and components - all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

power

subsystems

1

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents ~~[involving handling recently irradiated fuel]~~).

2

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of ~~[recently]~~ irradiated fuel assemblies, provide assurance that:

2

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core,

b. Systems needed to mitigate a fuel handling accident ~~[involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~ are available,

2

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

instrument

The AC, DC, and AC ~~vital~~ bus electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

1

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

BASES

ACTIONS (continued)

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of ~~[recently]~~ irradiated fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystems LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of ~~[recently]~~ irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required shutdown cooling (SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.~~4~~ do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the SDC ACTIONS would not be entered. Therefore, Required Action A.2.~~5~~ is provided to direct declaring SDC inoperable, which results in taking the appropriate SDC actions.

BASES

ACTIONS (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

instrument

This Surveillance verifies that the AC, DC, and AC ~~vital~~ bus electrical power distribution system is functioning properly, with all the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses.

1

~~[The 7 day Frequency takes into account the redundant capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

2

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

4

~~]~~

2

REFERENCES

1. FSAR, Chapter ~~[~~6~~]~~.

U

2. FSAR, Chapter ~~[~~15~~]~~.

2 1

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the AC, DC, and AC ~~vital~~ bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter ~~[~~6~~]~~ (Ref. 1) and Chapter ~~[~~15~~]~~ (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and AC ~~vital~~ bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC ~~vital~~ bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC, DC, and AC ~~vital~~ bus electrical power distribution subsystems during MODES 5 and 6, and during movement of ~~[recently]~~ irradiated fuel assemblies, ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods,

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident ~~[involving handling recently irradiated fuel. Due to radioactive decay, AC, DC, and AC vital bus electrical power is only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~.

The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific unit condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution ~~system~~ necessary to support OPERABILITY of required systems, equipment and components - all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

power

subsystems

1

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents ~~[involving handling recently irradiated fuel]~~).

2

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of ~~[recently]~~ irradiated fuel assemblies, provide assurance that:

2

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core,

b. Systems needed to mitigate a fuel handling accident ~~[involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days)]~~ are available,

2

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available, and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

instrument

The AC, DC, and AC ~~vital~~ bus electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

1

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

BASES

ACTIONS (continued)

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of ~~[recently]~~ irradiated fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystems LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of ~~[recently]~~ irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required shutdown cooling (SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.~~4~~ do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the SDC ACTIONS would not be entered. Therefore, Required Action A.2.~~5~~ is provided to direct declaring SDC inoperable, which results in taking the appropriate SDC actions.

BASES

ACTIONS (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

instrument

This Surveillance verifies that the AC, DC, and AC ~~vital~~ bus electrical power distribution system is functioning properly, with all the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses.

1

~~[The 7 day Frequency takes into account the redundant capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.~~

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

2

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

4

~~]~~

2

REFERENCES

1. FSAR, Chapter ~~[~~6~~]~~.

U

2. FSAR, Chapter ~~[~~15~~]~~.

2 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.10 BASES, DISTRIBUTION SYSTEMS - SHUTDOWN

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. Changes are made to the Bases to reflect the Specification.

4. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.8.10, DISTRIBUTION SYSTEMS – SHUTDOWN

There are no specific No Significant Hazards Considerations for this Specification.

ENCLOSURE 2

VOLUME 14

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.9
REFUELING OPERATIONS

Revision 1

LIST OF ATTACHMENTS

1. 3.9.1, Boron Concentration
2. 3.9.2, Nuclear Instrumentation
3. 3.9.3, Containment Penetrations
4. 3.9.4, Shutdown Cooling (SDC) and Coolant Circulation – High Water Level
5. 3.9.5, Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level
6. 3.9.6, Refueling Water Level
7. Relocated/Deleted Current Technical Specifications

ATTACHMENT 1

3.9.1, Boron Concentration

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

LCO 3.9.1

~~3.9.1~~ ~~With the reactor vessel head unbolted or removed~~, the boron concentration of ~~all filled portions of~~ the Reactor Coolant System and the refueling cavity shall be maintained within the limit specified in the COLR.

, the refueling canal,

A02

A03

Applicability

APPLICABILITY: MODE 6~~*~~.

A02

Add proposed Applicability Note

L01

ACTION:

Boron concentration not within limit

additions

ACTION A

ACTION A.1

ACTION A.2

~~With the requirements of the above specification not satisfied~~, immediately suspend ~~all operations involving CORE ALTERATIONS or~~ positive reactivity ~~changes~~ and initiate ~~and continue boration at ≥ 40 gpm of greater than or equal to 1900 ppm boron or its equivalent~~ to restore boron concentration to within limit~~s~~.

L02

L03

action

SURVEILLANCE REQUIREMENTS

~~4.9.1.1 The boron concentration limit shall be determined prior to:~~

~~a. Removing or unbolting the reactor vessel head, and~~

~~b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position.~~

L04

SR 3.9.1.1

~~4.9.1.2~~ The boron concentration ~~of the refueling cavity~~ shall be determined ~~by chemical analysis~~ in accordance with the Surveillance Frequency Control Program.

LA01

~~*~~ ~~The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.~~

A02

A01

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

LCO 3.9.1

~~3.9.1~~ ~~With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the~~ boron concentration of ~~all filled portions of~~ the Reactor Coolant System and the refueling cavity shall be maintained within the limit specified in the COLR.

A02

, the refueling canal,

A03

Applicability

APPLICABILITY: MODE 6~~*~~.

A02

Add proposed Applicability Note

L01

ACTION:

Boron concentration not within limit

ACTION A

ACTION A.1

ACTION A.2

~~With the requirements of the above specification not satisfied,~~ immediately suspend ~~all operations involving CORE ALTERATIONS or~~ positive reactivity ~~changes~~ and initiate ~~and continue boration at greater than or equal to 40 gpm of a solution containing 1900 ppm boron or greater~~ to restore boron concentration to within limit~~s~~.

L02

L03

additions

action

SURVEILLANCE REQUIREMENTS

~~4.9.1.1 The boron concentration limit shall be determined prior to:~~

~~a. Removing or unbolting the reactor vessel head, and~~

~~b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.~~

L04

SR 3.9.1.1

~~4.9.1.2~~ The boron concentration ~~of the reactor coolant system and the refueling canal~~ shall be determined ~~by chemical analysis~~ in accordance with the Surveillance Frequency Control Program.

LA01

~~*~~ ~~The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.~~

A02

DISCUSSION OF CHANGES
ITS 3.9.1, BORON CONCENTRATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.9.1 Applicability is MODE 6 with the footnote (*) reactor vessel head unbolted (i.e., not fully tensioned) or with the reactor vessel head removed. CTS 3.9.1 also includes this Applicability requirement in the LCO statement. ITS 3.9.1 requires the LCO in MODE 6. This changes the CTS by deleting the Applicability footnote and the LCO Applicability requirement.

CTS Table 1.2 defines MODE 6 with fuel in the reactor vessel and with the vessel head closure bolts less than fully tensioned (i.e., unbolted) or with the reactor vessel head removed. ITS Table 1.1-1 also defines MODE 6 with one or more reactor vessel head closure bolts less than fully tensioned, which encompasses vessel head removed. Additionally, the ITS definition of a MODE specifically defines a MODE, "…with fuel in the reactor vessel." Therefore, the CTS Applicability footnote and repeating the Applicability requirement in the LCO statement are not necessary because they duplicate requirements found in the definition.

This change is designated as an administrative change and is acceptable because it does not result in technical changes to the CTS.

A03 CTS 3.9.1 provides requirements on the boron concentration of all filled portions of the Reactor Coolant System (RCS) and the refueling cavity. ITS 3.9.1 provides requirements on the boron concentration of the RCS, the refueling canal, and the refueling cavity. This changes the CTS by explicitly including the refueling canal in the volumes required to have boron concentration maintained and removing the clarification related to all filled portions of the RCS. This change is acceptable because the technical requirements have not changed. It is inherently understood that the boron concentration requirement is related to the reactor coolant volume, and therefore, it is unnecessary to clarify "all filled portions." The refueling canal is considered an extension of the refueling cavity and contains the same water when connected to the RCS. Therefore, the refueling canal is governed by the CTS requirements because the refueling canal is connected to the refueling cavity during refueling operations, and therefore, connected to the RCS.

This change is designated as administrative because the technical requirements of the specifications have not changed.

DISCUSSION OF CHANGES
ITS 3.9.1, BORON CONCENTRATION

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* Unit 1 CTS 4.9.1.2 requires a determination of boron concentration of the refueling cavity "by chemical analysis" in accordance with the Surveillance Frequency Control Program. Unit 2 CTS 4.9.1.2 requires a determination of boron concentration of the reactor coolant system and the refueling canal "by chemical analysis" in accordance with the Surveillance Frequency Control Program. CTS 4.9.1.2 requires that the boron concentration of the Reactor Coolant System and the refueling canal be determined "by chemical analysis" in accordance with the Surveillance Frequency Control Program. ITS SR 3.9.1.1 requires verification that boron concentration is within the limit specified in the COLR. ITS SR 3.9.1.1 does not specify where boron concentration is to be determined and that the boron concentration be determined by chemical analysis. This changes the CTS by moving details of where and how the boron concentration is determined from the CTS to the Bases.

The removal of these procedural details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the boron concentration be verified within its limit. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Section 5.5. This program provides for the evaluation of changes to ensure the Bases are properly controlled.

This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.9.1 provides limits on the boron concentration of all filled portions of the Reactor Coolant System and the refueling cavity when in MODE 6. ITS 3.9.1 modifies this requirement with a Note which states "Only applicable to the refueling canal and refueling cavity when connected to the RCS." This changes the CTS by eliminating the

applicability of the boron concentration limits on the refueling canal and refueling cavity when those volumes are not connected to the RCS. In addition, ITS SR 3.9.1.1 requires a verification that the boron is within the limit specified in the COLR in accordance with the Surveillance Frequency Control Program.

The purpose of CTS 3.9.1 is to ensure the boron concentration of the water surrounding the reactor fuel is sufficient to maintain the required SHUTDOWN MARGIN. This change is acceptable because the requirements continue to ensure that process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. If the refueling canal and refueling cavity are not connected to the RCS (such as when the reactor vessel head is on the reactor vessel while de-tensioned), the boron concentration of those volumes cannot affect the reactor core SHUTDOWN MARGIN. In addition, the general requirements of ITS SR 3.0.1 (CTS 4.0.1) and SR 3.0.4 (CTS 4.0.4) ensure a boron concentration verification is performed prior to connecting the refueling canal and refueling cavity to the RCS, to ensure the newly connected portions cannot decrease the boron concentration below the limit.

This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* The CTS 3.9.1 Action specifies the compensatory action for when the boron concentration requirement is not met. One of the compensatory actions is to suspend CORE ALTERATIONS. Under similar conditions, ITS 3.9.1 does not require suspension of CORE ALTERATIONS. This changes the CTS by deleting the requirement to suspend CORE ALTERATIONS when the boron concentration requirement is not met.

The purpose of CTS 3.9.1 is to ensure the boron concentration of the water surrounding the reactor fuel is sufficient to maintain the required SHUTDOWN MARGIN. Thus, when the limit is not met, the CTS 3.9.1 Action suspends CORE ALTERATIONS to preclude an event that could result in not meeting the SHUTDOWN MARGIN limit. CORE ALTERATION is defined in CTS 1.9, in part, as "the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel." There are two evolutions encompassed under the term CORE ALTERATION that could affect the SHUTDOWN MARGIN: addition of fuel to the reactor vessel and withdrawal of control rods. However, ITS 3.9.1 Required Action A.1 requires immediate suspension of positive reactivity additions. This would include both the addition of fuel to the reactor vessel and the withdrawal of control rods. Furthermore, another accident considered in MODE 6 that could affect SHUTDOWN MARGIN is a boron dilution event. A boron dilution accident is initiated by a dilution source which results in the boron concentration dropping below that required to maintain the SHUTDOWN MARGIN. A boron dilution accident is mitigated by stopping the dilution. Suspension of CORE ALTERATIONS has no effect on the mitigation of a boron dilution accident.

Therefore, since the only CORE ALTERATIONS that could affect the SHUTDOWN MARGIN are suspended by ITS 3.9.1 Required Action A.1, deletion of the requirement to suspend CORE ALTERATIONS is acceptable.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L03 *(Category 4 – Relaxation of Required Action)* CTS 3.9.1 Action states that when the boron concentration requirement is not met, initiate, and continue boration at ≥ 40 gpm of a solution containing 1900 pm boron or greater to restore boron concentration to within limits. ITS 3.9.1 Required Action A.2 requires initiation of action to restore boron concentration to within limit. This changes the CTS by eliminating the specific requirements for the boric acid solution to be used to restore compliance with the LCO.

The purpose of CTS 3.9.1 Action is to restore the required SHUTDOWN MARGIN in a timely manner. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. Removing the specific values of flow rate and boron concentration from the CTS ACTION provides flexibility in the restoration of the SHUTDOWN MARGIN and eliminates conflicts between the SDM value specified in the COLR and the specific boration values in the CTS ACTION. As stated, in the ITS Bases for ACTION A:

> "In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions. Once boration is initiated, it must be continued until the boron concentration is restored."

Specifying a minimum flow rate and concentration in the ACTION may not accomplish the objective of raising the RCS boron concentration as soon as possible. Specifying the boric acid solution requirements in the Action is not necessary, since the ITS requires that action to restore the boron concentration to within limit be initiated immediately. This prompt action will result in the boron concentration being restored in a timely manner, consistent with the CTS requirement.

This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L04 *(Category 5 – Deletion of Surveillance Requirement*) CTS 4.9.1.1 requires the boron concentration to be determined prior to removing or unbolting the reactor vessel head, and prior to withdrawal of any control element assembly (CEA) in excess of 3 feet from its fully inserted position. ITS 3.9.1 does not contain this Surveillance Requirement.

The purpose of CTS 4.9.1.1 is to ensure that the LCO requirements are met prior to entering MODE 6 and that the reactor has sufficient SHUTDOWN MARGIN prior to withdrawing any CEAs. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the values used to meet the LCO are consistent with the safety analyses. Thus, appropriate values

continue to be tested in a manner and at a frequency necessary to give confidence that the assumptions in the safety analyses are protected. ITS 3.9.1 requires that the boron concentration be met in MODE 6 or that action be immediately initiated to restore the boron concentration and that all positive reactivity additions be suspended. CTS 4.0.4 (ITS SR 3.0.4) requires the SR to be met within its Frequency requirements prior to entering the MODE or other specified conditions in the Applicability except as provided by CTS 4.0.3 (ITS SR 3.0.3 – missed SR). Therefore, verification that the boron concentration is within the required limit must be performed prior to entering MODE 6 in order to avoid immediately entering into an Action that prohibits withdrawal of CEAs when the boron concentration requirement is not met. While the CTS Surveillance is not required, the level of protection provided is appropriate.

This change is designated as less restrictive because Surveillances required in the CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

3.9.1

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, ~~[~~the refueling canal, and the refueling cavity~~]~~ shall be maintained within the limit specified in the COLR.

2

Applicability

APPLICABILITY: MODE 6.

----------NOTE----------
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

DOC L01

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.1 Action DOC L02 DOC L03 A. Boron concentration not within limit.	A.1 Suspend positive reactivity additions. AND A.2 Initiate action to restore boron concentration to within limit.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.9.1.2
DOC L04

2

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

3.9.1

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, ~~[~~the refueling canal, and the refueling cavity~~]~~ shall be maintained within the limit specified in the COLR. 2

Applicability

APPLICABILITY: MODE 6.

----------NOTE----------
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

DOC L01

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend positive reactivity additions. AND A.2 Initiate action to restore boron concentration to within limit.	Immediately Immediately

3.9.1 Action
DOC L02
DOC L03

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	~~[72 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.9.1.2
DOC L04

2

DOC L02
DOC L03

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.1, BORON CONCENTRATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Unit procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control element assemblies (CEAs) and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water tank into the open reactor vessel by gravity feeding or by the use of the Shutdown Cooling (SDC) System pumps.

or other borated water source

1

The pumping action of the SDC System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and the refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The SDC System is in operation during refueling (see LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling and Coolant Circulation - Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

BASES

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in B 3.1.1, “SHUTDOWN MARGIN (SDM).”

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a k_{eff} ≤ 0.95. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensures that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the RCS. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

BASES

ACTIONS

A.1

Continuation of positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving positive reactivity additions must be suspended immediately. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

Suspension of positive reactivity additions shall not preclude moving a component to a safe position.

A.2

In addition to immediately suspending positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS, and connected portions of the refueling canal and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity or canal were disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section ~~[ ]~~. 15.2.4 2

U 1

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Unit procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control element assemblies (CEAs) and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water tank into the open reactor vessel by gravity feeding or by the use of the Shutdown Cooling (SDC) System pumps.

or other borated water source

1

The pumping action of the SDC System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and the refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The SDC System is in operation during refueling (see LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling and Coolant Circulation - Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

BASES

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in B 3.1.1, “SHUTDOWN MARGIN (SDM).”

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a k_{eff} ≤ 0.95. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensures that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the RCS. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

BASES

ACTIONS

A.1

Continuation of positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving positive reactivity additions must be suspended immediately. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

Suspension of positive reactivity additions shall not preclude moving a component to a safe position.

A.2

In addition to immediately suspending positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS, and connected portions of the refueling canal and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity or canal were disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section ~~[ ]~~. 15.4.6 2

U 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.1, BASES, BORON CONCENTRATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.1, BORON CONCENTRATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 2

3.9.2, Nuclear Instrumentation

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REFUELING OPERATIONS

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

source range | OPERABLE | A02

LCO 3.9.2 ~~3.9.2~~ ~~As a minimum,~~ two ~~wide range logarithmic neutron flux~~ monitors shall be ~~operating, each with continuous visual indication in the control room and one with audible indication in the containment~~.

LA01

~~control room~~ | M01

Applicability **APPLICABILITY:** MODE 6.

ACTION:

Add proposed ACTION A.2 | A03

Add proposed ACTION A.2 Note | L01

One required source range monitor inoperable.

~~With the requirements of the above specification not satisfied~~, immediately suspend ~~all~~ operations involving CORE ALTERATIONS ~~or operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1.~~ ~~The provisions of Specification 3.0.3 are not applicable.~~

ACTION A.2

ACTION A.1

A03

A04

Add proposed ACTION A.1 | M02

Add proposed ACTION B | M03

SURVEILLANCE REQUIREMENTS

4.9.2 ~~Each wide range logarithmic neutron flux monitor shall be demonstrated OPERABLE by performance of:~~

L02

~~a. A CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.~~

~~b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the start of CORE ALTERATIONS, and~~

SR 3.9.2.1

~~c.~~ ~~A~~ CHANNEL CHECK in accordance with the Surveillance Frequency Control Program ~~during CORE ALTERATIONS~~.

M05

Add proposed SR 3.9.2.2 and Note | M04

A01

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

source range

LCO 3.9.2 ~~3.9.2~~ ~~As a minimum,~~ two ~~startup range neutron flux~~ monitors shall be OPERABLE ~~and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.~~

A02 LA01 L03

Applicability **APPLICABILITY:** MODE 6.

ACTION:

Add proposed ACTION A.2 — A03

Add proposed ACTION A.2 Note — L01

source range

a. ~~With~~ one ~~of the above~~ required monitor~~s~~ inoperable ~~or not operating,~~ immediately suspend ~~all operations involving CORE ALTERATIONS~~ ~~or operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1.~~

ACTION A.2

ACTION A.1

A03

Add proposed ACTION A.1 — M02

two source range

ACTION B.2 b. ~~With both of the above~~ required monitors inoperable ~~or not operating,~~ determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

Add proposed ACTION B.1 — M03

c. ~~The provisions of Specification 3.0.3 are not applicable.~~ — A04

SURVEILLANCE REQUIREMENTS

~~4.9.2 Each startup range neutron flux monitor shall be demonstrated OPERABLE by performance of:~~

SR 3.9.2.1 a. ~~A~~ CHANNEL CHECK in accordance with the Surveillance Frequency Control Program,

L02

b. ~~A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and~~

c. ~~A CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.~~

Add proposed SR 3.9.2.2 and Note — M04

DISCUSSION OF CHANGES
ITS 3.9.2, NUCLEAR INSTRUMENTATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.9.2 requires, in part, that two neutron flux monitors are to be operating; wide range logarithmic (Unit 1) and startup range (Unit 2). Additionally, CTS 3.9.2 Actions contain compensatory actions to take when one or more neutron flux monitors are not operating. ITS LCO 3.9.2 requires, in part, two source range monitors (SRMs) to be OPERABLE but does not require the SRMs to be operating. Furthermore, ITS 3.9.2 ACTIONS A and B do not contain compensatory actions to take when one or more of the SRMs are not operating. This changes the CTS by removing the statement that the neutron flux monitors are required to be operating.

ITS refers to the neutron flux monitors as SRMs consistent with the ISTS because, for the purpose of CTS 3.9.2, both types of monitors (wide range logarithmic (Unit 1) and startup range (Unit 2)) provide neutron flux indication in the source range. The purpose of the SRMs is to monitor core reactivity during refueling operations and provide a signal to the operators if an unexpected reactivity change occurs. This change is acceptable because the requirements have not changed. In accordance with the ITS definition of OPERABLE, a device must be capable of performing its specified safety function. For the SRMs, this requires them to be operating in order to perform their safety function (i.e., monitor neutron flux in the source range). This change is considered administrative and acceptable because it does not result in a technical change to the CTS.

A03 Unit 1 CTS 3.9.2 Action and Unit 2 CTS Action a require, in part, with one neutron flux monitor inoperable to immediately suspend all operations involving CORE ALTERATIONS or immediately suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1, “Boron Concentration.” Under similar conditions, ITS 3.9.2 Required Action A.1 requires immediate suspension of positive reactivity additions. Additionally, ITS 3.9.2 Required Action A.2 requires immediate suspension of movement of fuel, sources, and reactivity control components within the reactor vessel. This changes the CTS by immediately suspending positive reactivity additions and movement of fuel, sources, and reactivity control components within the reactor vessel, instead of suspending CORE ALTERATIONS.

The purpose of the SRMs is to monitor core reactivity during refueling operations and provide a signal to the operators if an unexpected reactivity change occurs. Thus, when an SRM is inoperable, CORE ALTERATIONS are suspended to

preclude an unmonitored reactivity change. CORE ALTERATIONS is defined in CTS 1.9, in part, as "the movement of any fuel, sources, reactivity control components or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel." CORE ALTERATIONS only occur when the reactor vessel head is removed; therefore, it only applies to MODE 6. There are two evolutions encompassed under the term CORE ALTERATION that could affect the reactivity of the core; the addition of fuel to the reactor vessel and the withdrawal of control rods. ITS 3.9.2 Required Action A.1 requires immediate suspension of these two positive reactivity changes and prohibits diluting the boron concentration of the coolant in the RCS. Therefore, since the CORE ALTERATIONS of concern are only those that could affect positive reactivity in the core and these are suspended by ITS 3.9.2 Required Action A.1, changing the requirement from suspending "CORE ALTERATIONS" to suspending "positive reactivity additions" is acceptable.

ITS 3.9.2 Required Action A.2 prohibits the movement of fuel assemblies, sources, and reactivity control components when an SRM is inoperable. The proposed changes ensure that no actions are taken that could alter the core reactivity when an SRM is inoperable. With one SRM inoperable, the operator may not be able to monitor the core reactivity condition in part of the reactor. Therefore, the conservative action is to suspend movement of any core components that may affect reactivity until the SRM is restored. While unlikely, movement of fuel assemblies from one core location to another, the movement of sources, or the removal of reactivity control components, could result in an undetected change in core reactivity. These changes are considered administrative and acceptable because they do not result in a technical change to the CTS.

A04 Unit 1 CTS 3.9.2 Action and Unit 2 CTS 3.9.2 Action c. contain the statement, "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.2 does not contain an equivalent statement. This changes the CTS by deleting the Specification 3.0.3 exception. This change is acceptable because the technical requirements have not changed. As explicitly stated in CTS 3.0.3 (ITS LCO 3.0.3), LCO 3.0.3 is not applicable in MODE 6. Therefore, the CTS LCO 3.0.3 exception statement is not needed. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS 3.9.2 states that two wide range logarithmic neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication "in the containment." ITS 3.9.2 states that two source range monitors (SRMs) shall be OPERABLE. Details of what constitutes an OPERABLE SRM channel are provided in the ITS Bases (Refer to Discussion of Change (DOC) LA01). The ITS Bases explicitly states: To alert the operator of a boron dilution accident, at least one SRM must include audible indication in the control room to be considered OPERABLE. This changes the CTS by changing the location requirement for the audible indication from “the containment” to the “control room.”

This change is necessary to ensure that the SRM channels are maintained consistent with the safety analyses and licensing basis in MODE 6. The boron dilution analysis assumes that the operator has prompt and definite indication from an audible indication. The analysis does not explicitly specify how many audible channels are required or the location of the audible indication. However, action to terminate a boron dilution event is either initiated from an operator in the control room or directed by an operator in the control room. Therefore, it is more appropriate to specify that an audible indication be available in the control room instead of in the containment. This change is designated as more restrictive because a specific location of the audible alarm indication not previously required in the CTS is being applied in the ITS.

M02 Unit 1 CTS 3.9.1 Action and Unit 2 CTS 3.9.2 Action a require, in part, with one neutron flux monitor inoperable to, optionally, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1, "Boron Concentration." Under similar conditions, ITS 3.9.2 Required Action A.1 requires immediate suspension of positive reactivity additions. This changes the CTS requiring suspension of positive reactivity additions, which is not optional and includes RCS dilution activities even if the introduced coolant has a boron concentration greater than the limit in LCO 3.9.1.

The purpose of the source range flux monitoring requirements while in MODE 6 is to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. Pursuant to the requirements of 10 CFR 50.36(c)(2), the required action provides appropriate remedial actions when the required source range monitoring capability is degraded to ensure continued safe operation without adequate monitoring capability until the degraded condition can be corrected. The CTS Action that prohibits introducing coolant into the RCS unless that coolant has a boron concentration greater than or equal to the boron concentration limit in LCO 3.9.1 is deleted. This current action would allow dilution of the boron concentration in the RCS with one SRM inoperable provided the boron concentration is not reduced to less than the limit in LCO 3.9.1. This change is necessary to minimize the possibility of a boron dilution event when monitoring capability is unavailable or degraded by suspending any positive reactivity additions, including any RCS dilution activity suspended. This change is more restrictive because it requires an action that was optional in CTS and expands the action to include suspension of any boron dilution activity regardless of existing RCS boron concentration.

M03 Unit 1 CTS provides no action when two required wide range logarithmic neutron flux monitors are inoperable. Unit 2 CTS 3.9.2 Action b provides action for two inoperable required startup range neutron flux monitors, requiring determination of the Reactor Coolant System boron concentration at least once per 12 hours. ITS 3.9.2 ACTION B requires actions when two SRMs are inoperable. The ITS requires immediate initiation of action to restore one SRM to OPERABLE status and to perform a verification of boron concentration (per ITS SR 3.9.1.1) once per 12 hours. This changes the Unit 1 CTS requirements by requiring an additional verification of boron concentration every 12 hours when both SRMs

are inoperable, and changes Unit 1 and Unit 2 CTS by requiring an additional action to initiate immediate action to restore one SRM to OPERABLE status.

The purpose of this change is to provide necessary Required Actions that are appropriate for a possible condition that could be encountered. This change is acceptable because the proposed Required Actions are reasonable and necessary to ensure the reactor is maintained in a safe condition. This change is more restrictive because it provides for additional actions that the CTS does not require.

M04 CTS 4.9.2 requires the performance of a CHANNEL CHECK of the SRMs but does not require the performance of a CHANNEL CALIBRATION. ITS SR 3.9.2.2 requires performance of CHANNEL CALIBRATION of the SRMs at a periodic Frequency in accordance with the Surveillance Frequency Control Program (SFCP). This Surveillance is modified by a Note which states that the Neutron detectors are excluded from the CHANNEL CALIBRATION. This changes the CTS by adding a new Surveillance Requirement to periodically verify the calibration of the SRMs.

The purpose of ITS SR 3.9.2.2 is to provide additional assurance that the SRMs are capable of providing a reliable and accurate indication of core subcritical neutron flux. This change is acceptable because a CHANNEL CALIBRATION will continue to ensure OPERABILITY and proper operation of the SRMs in MODE 6. This Surveillance is modified by a Note which states that the Neutron detectors are excluded from the CHANNEL CALIBRATION.

PSL controls periodic Frequencies for Surveillances in accordance with the SFCP per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, SR 3.9.2.2 will be performed at a Frequency in accordance with the SFCP with an initial Frequency of 18 months consistent with the ISTS SR 3.9.2.2 and considers operating experience that these components usually pass the Surveillance when performed on an 18 month Frequency.

The SFCP was established as described in FPL (PSL Unit 1 and Unit 2) “Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program” (ADAMS Accession No. ML14070A087). The NRC issued Amendment No. 223 to Renewed Facility Operating License No. DPR-67 and Amendment No. 173 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie 1 and 2), respectively (ADAMS Accession No. ML15127A066).This change is more restrictive because it provides an additional Surveillance Requirement that was not required in the CTS.

M05 **Unit 1 only**: CTS 4.9.2.c requires a CHANNEL CHECK to be performed in accordance the Surveillance Frequency Control Program during CORE ALTERATIONS. ITS SR 3.9.2.1 requires a CHANNEL CHECK to be performed in accordance with the Surveillance Frequency Control Program. This changes the CTS by requiring the CHANNEL CHECK to be performed in accordance with the Surveillance Frequency Control Program even if CORE ALTERATIONS are not in progress.

The purpose of this change is to routinely verify the OPERABILITY of the neutron flux monitors in all conditions in which the LCO applies, not just during CORE ALTERATIONS. This change is acceptable because the Surveillance verifies OPERABILITY of both monitors to ensure the reactor is maintained in a safe condition. This change is more restrictive because it provides for additional testing that the CTS does not require.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.9.2 states, in part, that two source range neutron flux monitors shall be operating, "each with continuous visual indication in the control room." CTS also requires one neutron flux monitor to have audible indication; in the containment for Unit 1 and in the control room and containment for Unit 2. ITS 3.9.2 LCO states that two source range monitors (SRMs) shall be OPERABLE. This changes the CTS by moving the requirement that each channel has a continuous visual indication in the control room and one with audible indication from the CTS to the Bases.

The removal of this detail, which is related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement that two channels be OPERABLE and continues to require the associated Surveillance to verify OPERABILITY. The ITS Bases provides the description of the SRM detectors providing continuous visual indication and an audible alarm in the control room. The ITS Bases also includes a statement to ensure at least one SRM includes audible indication in the control room to be considered OPERABLE. Refer to other DOCs for CTS requirement specifying the location of the audible alarm indication. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 (*Category 4 – Relaxation of Required Action*) CTS 3.9.2 ACTION a requires, in part, with one neutron flux monitor inoperable to immediately suspend all operations involving CORE ALTERATIONS. Under similar conditions, ITS 3.9.2 Required Action A.2 requires immediate suspension of movement of fuel,

sources, and reactivity control components within the reactor vessel, which is equivalent to suspending CORE ALTERATIONS. Additionally, ITS 3.9.2 Required Action A.2 is modified by a Note that permits fuel assemblies, sources, and reactivity control components to be moved, if necessary, to restore an inoperable source range monitor or to complete movement of a component to a safe condition. This changes the CTS by allowing certain CORE ALTERATIONS to be continued in order to restore the SRM to OPERABLE status or complete movement of a component to a safe condition.

Pursuant to the requirements of 10 CFR 50.36(c)(2), the required action provides appropriate remedial actions when the required source range monitoring capability is degraded to ensure continued safe operation without adequate monitoring capability until the degraded condition can be corrected. The SRMs are located outside the reactor core. Troubleshooting, repair, or replacement of the inoperable source range neutron flux monitors may require moving fuel, sources, or reactivity control components away from the SRM location to minimize the radiation dose to the workers. As a result, suspension of activities specified in ITS 3.9.2, Required Action A.2 (i.e., CORE ALTERATIONS) with no exceptions has the adverse impact of precluding restoring the SRMs to OPERABLE status. This change is acceptable because the required actions continue to minimize actions that could result in reactivity changes within the core, while providing the ability to safely restore source range neutron monitoring capability. This change is consistent with the Note specified in ISTS Traveler TSTF-571-T, "Revise Actions for Inoperable Source Range Neutron Flux Monitor," Revision 2, dated April 13, 2000. ISTS Traveler TSTF-571-T addressed, in part, NRC concerns with the adoption of ISTS Traveler TSTF-286 as documented in a letter to the TSTF dated October 4, 2018 (NRC ADAMS Accession No. ML17346A587). ISTS Traveler TSTF-571-T was incorporated into Revision 5 of the ISTS as recommended by the NRC staff. The October 4, 2018, letter states, in part, that "If a licensee includes the changes of traveler TSTF-571-T when adopting TSTF-286, the NRC staff's technical concerns should be adequately addressed with regard to TSTF-286. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than applied in the CTS.

L02 *(Category 5 – Deletion of Surveillance Requirement)* Unit 1 CTS 4.9.2.a and b, and Unit 2 CTS 4.9.2.b and c, state that a CHANNEL FUNCTIONAL TEST is required for the neutron flux monitors in accordance with the Surveillance Frequency Control Program and within 8 hours prior to the initial start of CORE ALTERATIONS, respectively. ITS 3.9.2 does not require the performance of similar tests for the required neutron flux monitors. This changes the CTS by deleting the CHANNEL FUNCTIONAL TEST Frequency of within 8 hours of CORE ALTERATIONS.

This change is acceptable because the deleted Surveillance Requirement Frequency is not necessary to verify that the equipment used to meet the LCO is consistent with the safety analysis. The SRMs continue to be tested in a manner and at a frequency necessary to give confidence that the assumptions in the safety analyses are protected. This change is designated as less restrictive because a Surveillance required in the CTS will not be required in the ITS.

L03 **Unit 2 only:** (*Category 1 – Relaxation of LCO Requirements*) CTS 3.9.2 requires two startup range neutron flux monitors to be operating, each with continuous visual indication in the control room and one with audible indication "in the containment and the control room." ITS 3.9.2 states that two source range monitors (SRMs) shall be OPERABLE. Details of what is constitutes an OPERABLE SRM channel are provided in the ITS Bases (Refer to DOC LA01). The ITS Bases explicitly states: To alert the operator of a boron dilution accident, at least one SRM must include audible indication in the control room to be considered OPERABLE. This changes the CTS by deleting the requirement for an audible indication to be "in the containment."

This change is acceptable because the LCO requirements continue to ensure that the SRM channels are maintained consistent with the safety analyses and licensing basis in MODE 6. The boron dilution analysis assumes that the operator has prompt and definite indication from an audible indication. The analysis does not explicitly specify how many audible channels are required or the location of the audible indication. However, action to terminate a boron dilution event is either initiated from an operator in the control room or directed by an operator in the control room. Therefore, it is unnecessary to specify that an audible indication be available in the containment. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

3.9.2 DOC M01 DOC LA01

LCO 3.9.2 Two source range monitors (SRMs) shall be OPERABLE. 1

Applicability

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
3.9.2 Action DOC M02 A. One ~~[~~required~~]~~ SRM inoperable.	A.1	Suspend positive reactivity additions.	Immediately 1 2
	AND		
DOC L01	A.2	---------- NOTE ---------- Fuel assemblies, sources, and reactivity control components may be moved if necessary to restore an inoperable source range neutron flux monitor or to complete movement of a component to a safe condition. ---------- 1	
DOC A03		Suspend movement of fuel, sources, and reactivity control components within the reactor vessel.	Immediately 1
DOC M03 B. Two ~~[~~required~~]~~ SRMs inoperable.	B.1	Initiate action to restore one SRM to OPERABLE status.	Immediately 1 2
	AND		
DOC M03	B.2	Perform SR 3.9.1.1.	Once per 12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.9.2.a DOC M05	SR 3.9.2.1	Perform CHANNEL CHECK.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
DOC M04	SR 3.9.2.2	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

3.9.2 DOC L03 DOC LA01

LCO 3.9.2 Two source range monitors (SRMs) shall be OPERABLE. 1

Applicability

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
3.9.2 Action a. DOC M02 A. One ~~[~~required~~]~~ SRM inoperable.	A.1	Suspend positive reactivity additions.	Immediately	1, 2
	AND			
DOC L01	A.2	---------- NOTE ---------- Fuel assemblies, sources, and reactivity control components may be moved if necessary to restore an inoperable source range neutron flux monitor or to complete movement of a component to a safe condition. ----------		1
DOC A03		Suspend movement of fuel, sources, and reactivity control components within the reactor vessel.	Immediately	1
DOC M03 B. Two ~~[~~required~~]~~ SRMs inoperable.	B.1	Initiate action to restore one SRM to OPERABLE status.	Immediately	1, 2
	AND			
3.9.2 Action b.	B.2	Perform SR 3.9.1.1.	Once per 12 hours	

1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.9.2.c	SR 3.9.2.1	Perform CHANNEL CHECK.	[12 hours OR In accordance with the Surveillance Frequency Control Program]	2 2
DOC M04	SR 3.9.2.2	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	[[18] months OR In accordance with the Surveillance Frequency Control Program]	2 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.2, NUCLEAR INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range monitors (SRMs) are used during refueling operations to monitor the core reactivity condition. The installed SRMs are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors is permitted, provided the LCO requirements are met.

include wide range logarithmic gamma-metric

The installed SRMs ~~are BF3~~ detectors ~~operating in the proportional region of the gas filled detector characteristic curve~~. ~~The~~ detectors monitor the neutron flux in counts per second. ~~The instrument range covers five decades of neutron flux (1E+5 cps) with a [5%] instrument accuracy~~. The detectors ~~also~~ provide continuous visual indication ~~in the control room~~ and an audible alarm to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

In the source range, the

(cps). The instrument range of the wide range logarithmic detectors is 1 to 10^5 cps, each

in the control room

or other neutron flux detectors

If used, portable detectors should be functionally equivalent to the ~~NIS~~ SRMs.

described herein

1

APPLICABLE SAFETY ANALYSES

Two OPERABLE SRMs are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that normally available SHUTDOWN MARGIN would be reduced, but there is sufficient time for the operator to take corrective actions.

The SRMs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires two SRMs OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

1

APPLICABILITY

In MODE 6, the SRMs must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels.

logarithmic fission

In MODES ~~2,~~ 3, 4, and 5, the installed ~~source range~~ detectors and circuitry are required to be OPERABLE by ~~LCO 3.3.2, "RPS Instrumentation Shutdown."~~

LCO 3.3.11, "Logarithmic Neutron Flux Monitor."

1

To alert the operator of a boron dilution accident, at least one SRM must include audible indication in the control room to be considered OPERABLE.

BASES

ACTIONS

A.1 and A.2

of visual indication

With only one SRM OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, positive reactivity additions and movement of fuel, sources, and reactivity control components within the reactor vessel must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position. Suspending the movement of fuel, sources, and reactivity control components ensures that positive reactivity is not inadvertently added to the reactor core while the SRM is inoperable. Required Action A.2 is modified by a Note that states that fuel assemblies, sources, and reactivity control components may be moved if necessary to facilitate repair or replacement of the inoperable SRM. It may be necessary to move these items away from the locations in the core close to the SRM to minimize personnel radiation dose during troubleshooting or repair. The Note also permits completion of movement of a component to a safe position, should the SRM be discovered inoperable during component movement.

B.1

With no SRM OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until an SRM is restored to OPERABLE status.

B.2

and no audible indication in the control room to detect a boron dilution event

With no SRM OPERABLE, there is no direct means of detecting changes in core reactivity. However, since positive reactivity additions are not to be made, the core reactivity condition is stabilized until the SRMs are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to verify that the required boron concentration exists.

, including boron dilution activities

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this period.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

~~[The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1, "Reactor Protection System."~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. ~~[The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
2. U FSAR, Section 15.2.4 ~~[ ]~~. 1 2

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range monitors (SRMs) are used during refueling operations to monitor the core reactivity condition. The installed SRMs are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors is permitted, provided the LCO requirements are met.

include excore startup gamma-metric fission

The installed SRMs ~~are BF3~~ detectors ~~operating in the proportional region of the gas filled detector characteristic curve~~. The detectors monitor the neutron flux in counts per second. ~~The instrument range covers five decades of neutron flux (1E+5 cps) with a [5%] instrument accuracy.~~ The detectors ~~also~~ provide continuous visual indication ~~in the control room~~ and an audible alarm to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

(cps). The instrument range of the excore startup detectors is 1 to 10^5 cps.

in the control room

or other neutron flux detectors

If used, portable detectors should be functionally equivalent to the ~~NIS~~ SRMs.

described herein

1

APPLICABLE SAFETY ANALYSES

Two OPERABLE SRMs are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that normally available SHUTDOWN MARGIN would be reduced, but there is sufficient time for the operator to take corrective actions.

The SRMs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires two SRMs OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

1

APPLICABILITY

In MODE 6, the SRMs must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels.

logarithmic fission

In MODES ~~2,~~ 3, 4, and 5, the installed ~~source range~~ detectors and circuitry are required to be OPERABLE by ~~LCO 3.3.2, "RPS Instrumentation Shutdown."~~

LCO 3.3.11, "Logarithmic Neutron Flux Monitor."

To alert the operator of a boron dilution accident, at least one SRM must include audible indication in the control room to be considered OPERABLE.

1

1

BASES

ACTIONS

A.1 and A.2

of visual indication

With only one SRM OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, positive reactivity additions and movement of fuel, sources, and reactivity control components within the reactor vessel must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position. Suspending the movement of fuel, sources, and reactivity control components ensures that positive reactivity is not inadvertently added to the reactor core while the SRM is inoperable. Required Action A.2 is modified by a Note that states that fuel assemblies, sources, and reactivity control components may be moved if necessary to facilitate repair or replacement of the inoperable SRM. It may be necessary to move these items away from the locations in the core close to the SRM to minimize personnel radiation dose during troubleshooting or repair. The Note also permits completion of movement of a component to a safe position, should the SRM be discovered inoperable during component movement.

1

B.1

With no SRM OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until an SRM is restored to OPERABLE status.

B.2

and no audible indication in the control room to detect a boron dilution event

With no SRM OPERABLE, there is no direct means of detecting changes in core reactivity. However, since positive reactivity additions are not to be made, the core reactivity condition is stabilized until the SRMs are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to verify that the required boron concentration exists.

, including boron dilution activities

1

1

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this period.

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

~~[The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1, "Reactor Protection System."~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. ~~[The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

~~]~~ 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
2. U FSAR, Section 15.4.2 ~~[ ]~~. 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.2, BASES, NUCLEAR INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.2, NUCLEAR INSTRUMENTATION

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 3

3.9.3, Containment Penetrations

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REFUELING OPERATIONS

CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

LCO 3.9.3

~~3.9.4~~ The containment penetrations shall be in the following status:

hatch

3.9.3.a

~~a.~~ The equipment door closed and held in place by ~~a minimum of~~ four bolts.

3.9.3.b

~~b.~~ ~~A minimum of~~ one door in each airlock is closed.

3.9.3.c

~~c.~~ Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

a manual or automatic

3.9.3.c.1

~~1.~~ Closed by isolation valve, blind flange~~, or manual valve except for valves that are open on an intermittent basis under administrative control~~, or

A03

3.9.3.c.2

~~2.~~ ~~Be~~ capable of being closed by an OPERABLE ~~automatic~~ containment isolation valve, or

~~3. Be capable of being closed by an OPERABLE containment vacuum relief valve.~~

A04

3.9.3 Note

Note: Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

Applicability

APPLICABILITY: During movement of recently irradiated fuel within the containment.

ACTION:

One or more containment penetrations not in required status.

assemblies within

ACTION A

~~With the requirements of the above specification not satisfied~~, immediately suspend ~~all operations involving~~ movement of recently irradiated fuel ~~in the~~ containment. ~~The provisions of Specification 3.0.3 are not applicable.~~

A05

SURVEILLANCE REQUIREMENTS

is in the required status

SR 3.9.3.1 & SR 3.9.3.2

~~4.9.4~~ Each ~~of the above~~ required containment penetration~~s shall be determined to be~~ either in its closed/isolated condition or capable of being closed by an OPERABLE ~~automatic~~ containment isolation valve~~ within 72 hours prior to the start of and~~ in accordance with the Surveillance Frequency Control Program ~~during movement of recently irradiated fuel in the containment~~ by:

L01

A02

SR 3.9.3.1 & SR 3.9.3.2 Frequency

SR 3.9.3.1

a. Verify~~ing the penetrations are in their closed/isolated condition~~, or

each required containment is in the required status

SR 3.9.3.2

b. ~~Testing of containment isolation valves per the applicable portions of Specifications 4.6.3.1.1. and 4.6.3.1.2~~.

Add proposed SR 3.9.3.2 Note

L02

Verify each required containment isolation valve actuates to the isolation position on an actual or simulated actuation signal.

L03

~~INTENTIONALLY DELETED~~

~~INTENTIONALLY DELETED~~

~~**INTENTIONALLY DELETED**~~

A01

REFUELING OPERATIONS

CONTAINMENT ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.9.3.c

~~3.9.9~~ The containment isolation system shall be OPERABLE.

Applicability

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTION:

LCO 3.9.3.c.1
ACTION A.1

With the containment isolation system inoperable, ~~either~~ suspend ~~all operations involving~~ movement of recently irradiated fuel assemblies within containment ~~or close each of the penetrations providing direct access from the containment atmosphere to the outside atmosphere~~.

A06

SURVEILLANCE REQUIREMENTS

SR 3.9.3.2

~~4.9.9~~ The containment isolation system shall be demonstrated OPERABLE ~~within 72 hours prior to the start of and~~ in accordance with the Surveillance Frequency Control Program ~~during movement of recently irradiated fuel assemblies~~ by verifying that containment isolation occurs ~~on manual initiation and on a high radiation signal from two of the containment radiation monitoring instrumentation channels~~.

L01

A02

LA01

A01

REFUELING OPERATIONS

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

LCO 3.9.3

~~3.9.4~~ The containment ~~building~~ penetrations shall be in the following status:

hatch

3.9.3.a

~~a.~~ The equipment door closed and held in place by ~~a minimum of~~ four bolts.

3.9.3.b

~~b.~~ ~~A minimum of~~ one door in each airlock is closed.

3.9.3.c

~~c.~~ Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

manual or automatic

3.9.3.c.1

~~1.~~ Closed by a~~n~~ isolation valve, blind flange~~, or manual valve~~, or

3.9.3.c.2

~~2.~~ ~~Be~~ capable of being closed by an OPERABLE ~~automatic~~ containment isolation valve.

3.9.3 Note

Note: Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

Applicability

APPLICABILITY: During movement of recently irradiated fuel within ~~the~~ containment.

ACTION:

One or more containment penetrations not in required status.

ACTION A

~~With the requirements of the above specification not satisfied~~, immediately suspend ~~all operations involving~~ movement of recently irradiated fuel ~~in the~~ containment ~~building~~.

assemblies within

SURVEILLANCE REQUIREMENTS

is in the required status

SR 3.9.3.1 & SR 3.9.3.2

~~4.9.4~~ Each ~~of the above~~ required containment building penetration~~s~~ ~~shall be determined to be~~ either in its closed/isolated condition or capable of being closed by an OPERABLE ~~automatic~~ containment isolation valve ~~within 72 hours prior to the start of and~~ in accordance with the Surveillance Frequency Control Program ~~during movement of recently irradiated fuel in the containment building~~ by:

L01

A02

SR 3.9.3.1 & SR 3.9.3.2 Frequency

SR 3.9.3.1

~~a.~~ Verify~~ing the penetrations are in their closed/isolated condition~~, or

each required containment is in the required status

SR 3.9.3.2

~~b.~~ ~~Testing of containment isolation valves per the applicable portions of Specification 4.6.3.2~~.

Add proposed SR 3.9.3.2 Note

L02

Verify each required containment isolation valve actuates to the isolation position on an actual or simulated actuation signal.

L03

~~INTENTIONALLY DELETED~~

~~INTENTIONALLY DELETED~~

~~**INTENTIONALLY DELETED**~~

A01

REFUELING OPERATIONS

3/4.9.9 CONTAINMENT ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.9.3.c

~~3.9.9~~ The containment isolation system shall be OPERABLE.

Applicability

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTION:

LCO 3.9.3.c.1
ACTION A.1

With the containment isolation system inoperable, ~~either~~ suspend ~~all operations involving~~ movement of recently irradiated fuel assemblies within containment ~~or close each of the containment penetrations providing direct access from the containment atmosphere to the outside atmosphere~~.

A06

SURVEILLANCE REQUIREMENTS

SR 3.9.3.2

~~4.9.9~~ The containment isolation system shall be demonstrated OPERABLE ~~within 72 hours prior to the start of and~~ in accordance with the Surveillance Frequency Control Program ~~during movement of recently irradiated fuel~~ by verifying that containment isolation occurs ~~on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels~~.

L01

A02

LA01

DISCUSSION OF CHANGES
ITS 3.9.3, CONTAINMENT PENETRATIONS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 4.9.4 requires that each required containment penetration is in its required position in accordance with the Surveillance Frequency Control Program during movement of recently irradiated fuel assemblies within containment. CTS 4.9.9 requires that the containment isolation system be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during movement of recently irradiated fuel assemblies within containment. ITS SR 3.9.3.1 also requires verification that that each required containment penetration is in its required position in accordance with the Surveillance Frequency Control Program. The CTS and ITS applicability is also during movement of recently irradiated fuel assemblies within containment. This changes the CTS by deleting the Applicability requirement from the Surveillance. ITS LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification). Additionally, SR 3.0.1 establishes the requirement that Surveillance Requirements must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. Therefore, maintaining the Applicability requirement in the Surveillance is not necessary.

This change is designated as an administrative change and is acceptable because it does not result in technical changes to the CTS.

A03 **Unit 1 only:** CTS 3.9.4.c.1 states, in part, “except for valves that are open on an intermittent basis under administrative control.” CTS 3.9.4 Note provides this exception. ITS 3.9.3 Note provides this exception. This changes the CTS by deleting the CTS 3.9.4.c.1 statement “except for valves that are open on an intermittent basis under administrative control.”

This change is designated as administrative because it does not result in a technical change to the CTS.

A04 **Unit 1 only:** CTS 3.9.4.c.1 and c.3 state that each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed by isolation valve, blind flange, or manual valve (c.1), or be capable of being closed by an OPERABLE containment vacuum relief valve (c.3). Similarly, ITS 3.9.3 c.1 requires each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed by a manual or automatic isolation valve, blind flange, or equivalent, however; ITS does not

contain a specific requirement for the containment vacuum relief valve. This changes the CTS by deleting the explicit requirement for the containment vacuum relief valve. This change is acceptable because a containment vacuum relief valve is normally closed and automatically opens using a pneumatic operator under certain conditions. The valves automatically open and allow free air flow across the containment vessel boundary whenever a vacuum condition inside containment threatens vessel integrity. Since the vacuum relief valves also perform a containment isolation function in the event of a LOCA, the valves fail closed on loss of air. CTS and ITS both allow containment penetrations to be closed by a manual or automatic isolation valve. Therefore, it is not necessary to retain the explicit requirement for the containment vacuum relief valves because for the purpose of containment penetration Specifications, these valves are considered containment isolation valves.

This change is designated as administrative because it does not result in a technical change to the CTS.

A05 **Unit 1 only:** CTS 3.9.4 states "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.3 does not include this statement. This changes the CTS by deleting the Specification 3.0.3 exception. This change is acceptable because LCO 3.0.3 is not applicable in MODE 6. Therefore, the CTS 3.0.3 exception is not necessary.

This change is designated as administrative because it does not result in a technical change to the CTS.

A06 CTS 3.9.9 requires the containment isolation system to be OPERABLE during movement of recently irradiated fuel assemblies within containment. CTS 3.9.9 Action requires either suspending movement of recently irradiated fuel assemblies within containment or close each of the penetrations providing direct access from the containment atmosphere to the outside atmosphere. ITS 3.9.3 ACTIONS do not include the action to close each of the penetrations providing direct access from the containment atmosphere to the outside atmosphere. This changes the CTS by removing an unnecessary action. The purpose of the CTS 3.9.9 actions is to provide appropriate remedial actions when an LCO is not met consistent with the requirements of 10 CFR 50.36(c)(2)(i). The action to suspend movement of recently irradiated fuel assemblies within containment results in the LCO no longer being required. CTS 3.0.2 (ITS LCO 3.0.2) states, in part, that when an LCO is met or is no longer applicable prior to expiration of the specified time interval(s), completion of the ACTIONS is not required, unless otherwise stated. As indicated in the ISTS Bases of ITS LCO 3.0.2, “Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.” Therefore, it is unnecessary to provide an action to specifically require the LCO to be met, as this is inherent in LCO 3.0.1 and LCO 3.0.2.

This change is designated as an administrative change and is acceptable because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 4.9.9 requires a verification that containment isolation occurs on manual initiation and on a high radiation signal from two of the containment radiation monitoring instrumentation channels. ITS SR 3.9.3.2 requires verifying each required containment isolation valve actuates to the isolation position on an actual or simulated actuation signal. This changes the CTS by moving the details of the actuation signal that isolates the required containment isolation valves to the ITS Bases.

The removal of these details, which are related to system actuation design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The Technical Specifications still retain the surveillance requirement to verify the required containment isolation valves close from the applicable actuation instrumentation. This change is acceptable because the removed information will be adequately controlled in the Technical Specification Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Section 5.5 of the Technical Specifications. This program provides for the evaluation of Bases changes in accordance with 10 CFR 50.59 to ensure the Bases are properly controlled. The ITS requirement is consistent with the ISTS wording for this requirement. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.9.4 requires a determination that each containment penetration is either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment isolation valve within 72 hours prior to the start of movement of recently irradiated fuel assemblies in containment and in accordance with the Surveillance Frequency Control Program (i.e., 7 days). CTS 4.9.9 requires that the containment isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and in accordance with the Surveillance Frequency Control Program (i.e., 7 days). ITS SR 3.9.3.1 requires verification that each required containment penetration is in the required status in accordance with the Surveillance Frequency Control Program (i.e., 7 days). This changes the CTS by reducing the Frequency for verifying the status of the containment penetrations

and the containment isolation system from 72 hours before entering the Applicability of the LCO to 7 days before entering the Applicability of the LCO.

The purpose of CTS 4.9.4 and 4.9.9 is to verify that the valves associated with containment penetrations providing direct access to the outside atmosphere are in the correct position or the containment isolation valves are capable of being closed by an actuation signal to support handling or recently irradiated fuel assemblies. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that the containment penetrations are verified on a periodicity commensurate with the normal duration of time to complete fuel handling operations such that the verification will occur two or three times during the applicable period of the Specification, including once prior to movement of recently irradiated fuel. The Frequency of 7 days is sufficient during the movement of recently irradiated fuel assemblies; and therefore, it is sufficient before recently irradiated fuel assemblies are moved. CTS 4.0.4 (ITS SR 3.0.4) requires the SR to be met within its Frequency requirements prior to entering the MODE or other specified conditions in the Applicability except as provided by CTS 4.0.3 (ITS SR 3.0.3 – missed SR). The containment penetrations must be in the required state prior to commencing movement of recently irradiated fuel assemblies or fuel assembly movement must be suspended immediately (thereby exiting the Applicability of the Specification). Additionally, it is improbable to move irradiated fuel within 72 hours from entering MODE 3 (i.e., keff < 0.99) because of the physical time required to perform plant shutdown, cooldown, depressurize the Reactor Coolant System, and the additional operations required prior to moving recently irradiated fuel in the reactor vessel (e.g., containment entry, removal of vessel head, removal of vessel internals, etc.). Therefore, changing the Frequency from 72 hours before moving recently irradiated fuel assemblies to within 7 days before moving recently irradiated fuel assemblies has no effect on plant safety. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L02 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 4.9.4.b does not provide an exception to performance of the Surveillance for containment isolation valve(s) in penetrations closed to comply with the LCO 3.9.4.c.1. ITS SR 3.9.3.2 is modified by a Note that states that SR 3.9.3.2 is not required to be met for containment isolation valve(s) in penetrations closed to comply with ITS LCO 3.9.3.c.1. This changes the CTS by adding a Note that the SR is not required to be met for containment ventilation isolation valve(s) in penetrations closed to comply with LCO 3.9.4.c.1.

The purpose of CTS 4.9.4.b is to verify the equipment required to meet the LCO is OPERABLE. This change is acceptable because when containment ventilation isolation valves are closed to comply with ITS LCO 3.9.3.c.1, the penetrations are in the expected condition (isolated) to mitigate the effects of a fuel handling accident inside containment. Therefore, it is not necessary for the actuation signal to reposition valves to the closed position. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L03 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)*
Unit 1 CTS 4.9.4 requires testing containment isolation valves per the applicable portions of 4.6.3.1.1 and 4.6.3.1.2. Unit 2 CTS 4.9.4 requires testing containment isolation valves per the applicable portions of 4.6.3.2. Unit 1 CTS 4.6.3.1.1 and Unit 2 CTS 4.9.4 require isolation valves shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of the cycling test, and verification of isolation time. CTS 4.6.3.1.2 (Unit 1) and 4.6.3.2 (Unit 2) require, in part, verification that on a Containment Isolation (and Containment High Radiation – Unit 2) test signal, the isolation valve actuates to its isolation position. ITS SR 3.9.3.2 specifies that the signal may be from either an "actual" or simulated (i.e., test) signal. Also, Unit 1 ITS SR 3.9.3.2 does not include the CTS 4.6.3.1.1 Frequency. This changes the Unit 1 and Unit 2 CTS by explicitly allowing the use of either an actual or simulated signal for the test and also changes Unit 1 CTS by deleting the requirement to perform the Surveillance prior to return the valve to service following certain activities per CTS 4.6.3.1.1.

The purpose of CTS 4.9.4 is to ensure that the required containment isolation valves operate correctly upon receipt of an actuation signal. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Equipment cannot discriminate between an "actual," "simulated," or "test" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change allows taking credit for unplanned actuation if sufficient information is collected to satisfy the Surveillance test requirements. The change also allows a simulated signal to be used, if necessary. Additionally, Unit 1 CTS 4.6.3.1.1 and Unit 2 CTS 4.9.4 are proposed to be deleted in ITS 3.6.3 and thus, is not addressed here (Refer to Discussion of Change L03 associated with ITS 3.6.3, Containment Isolation Valves). This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

3.9.4 LCO 3.9.3 The containment penetrations shall be in the following status:

3.9.4.a a. The equipment hatch closed and held in place by ~~[~~four~~]~~ bolts, 2

3.9.4.b b. One door in each air lock is ~~[capable of being]~~ closed, and 2

3.9.4.c c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

3.9.4.c.1 1. Closed by a manual or automatic isolation valve, blind flange, or ~~equivalent or~~ 1

3.9.4.c.2 2. Capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Isolation ~~System~~ valve. 1

3.9.4 Note ----------NOTE----------
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

Applicability APPLICABILITY: During movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.4 Action A. One or more containment penetrations not in required status.	A.1 Suspend movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment.	Immediately

2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.9.4.a DOC L01	SR 3.9.3.1	Verify each required containment penetration is in the required status.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.9.4.b DOC L02 DOC L03	SR 3.9.3.2	----------NOTE---------- isolation Not required to be met for containment ~~purge and exhaust~~ valve(s) in penetrations closed to comply with LCO 3.9.3.c.1. ---------- Verify each required containment ~~purge and~~ isolation ~~exhaust~~ valve actuates to the isolation position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

3.9.4 LCO 3.9.3 The containment penetrations shall be in the following status:

3.9.4.a a. The equipment hatch closed and held in place by ~~[~~four~~]~~ bolts, 2

3.9.4.b b. One door in each air lock is ~~[capable of being]~~ closed, and 2

3.9.4.c c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

3.9.4.c.1 1. Closed by a manual or automatic isolation valve, blind flange, or ~~equivalent or~~ 1

3.9.4.c.2 2. Capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Isolation ~~System~~ valve. 1

3.9.4 Note ----------NOTE----------
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

Applicability APPLICABILITY: During movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.4 Action A. One or more containment penetrations not in required status.	A.1 Suspend movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment.	Immediately

2

1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.9.4.a DOC L01	SR 3.9.3.1	Verify each required containment penetration is in the required status.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2
4.9.4.b DOC L02 DOC L03	SR 3.9.3.2	----------NOTE---------- isolation ---- Not required to be met for containment ~~purge and exhaust~~ valve(s) in penetrations closed to comply with LCO 3.9.3.c.1. ---------- Verify each required containment ~~purge and exhaust~~ isolation valve actuates to the isolation position on an actual or simulated actuation signal.	~~[[18] months~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.3, CONTAINMENT PENETRATIONS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment, a release of fission product radioactivity within the containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR ~~100~~ 50.67. Additionally, the containment structure provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must ~~always~~ remain ~~[capable of being]~~ closed.

BASES

BACKGROUND (continued)

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

INSERT 1

~~The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 42 inch purge penetration and a 42 inch exhaust penetration. The second subsystem, a minipurge system, includes an 8 inch purge penetration and an 8 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and exhaust penetrations are secured in the closed position. The two valves in each of the two minipurge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.~~

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~~In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 42 inch purge system is used for this purpose and all valves are closed by the ESFAS in accordance with LCO 3.3.2, "Reactor Protective System (RPS) - Shutdown."~~

~~[The minipurge system remains operational in MODE 6 and all four valves are also closed by the ESFAS.~~

~~[or]~~

~~The minipurge system is not used in MODE 6. All four [8] inch valves are secured in the closed position.]~~

The ~~other~~ containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated ~~on at least one side~~. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during ~~[~~recently~~]~~ irradiated fuel movements ~~(Ref. 1)~~.

1

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1

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident ~~[involving handling recently irradiated fuel]~~. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. ~~2~~). Fuel handling accidents, ~~analyzed in Ref. 3, include~~ dropping a

2

1

1

involve

1

1

INSERT 1

The Containment Purge and Exhaust System includes 48 inch purge supply and exhaust penetrations, and a second subsystem, the Hydrogen Purge System, includes a 2 inch makeup penetration (containing two manual isolation valves) and a 2 inch exhaust penetration (containing two 3 inch air operated isolation valves). During MODES 1, 2, 3, and 4, the two 48 inch valves in the purge supply and exhaust penetrations are secured in the closed position. The 2 inch and 3 inch valves can be opened intermittently. The purge supply and exhaust valves (48 inch and 3 inch) are closed automatically by the containment isolation signal (CIS).

BASES

APPLICABLE SAFETY ANALYSES (continued)

INSERT 2

single irradiated fuel assembly ~~and handling tool or a heavy object onto other irradiated fuel assemblies.~~ ~~The requirements of LCO 3.9.6, "Refueling Water Level," in conjunction with minimum decay time of [72] hours prior to [irradiated fuel movement with containment closure capability or a minimum decay time of [x] days without containment closure capability], ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. The acceptance limits for offsite radiation exposure are contained in Standard Review Plan Section 15.7.4, Rev. 1 (Ref. 3), which defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values.~~ 1

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

~~----------REVIEWER'S NOTE----------~~
~~The allowance to have containment personnel air lock doors open and penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement is based on (1) confirmatory dose calculations of a fuel handling accident as approved by the NRC staff which indicate acceptable radiological consequences and (2) commitments from the licensee to implement acceptable administrative procedures that ensure in the event of a refueling accident (even though the containment fission product control function is not required to meet acceptable dose consequences) that the open air lock can and will be promptly closed following containment evacuation and that the open penetration(s) can and will be promptly closed. The time to close penetrations or combination of penetrations shall be included in the confirmatory dose calculations.~~
~~---------- 3~~

This LCO limits the consequences of a fuel handling accident ~~[~~involving handling recently irradiated fuel~~]~~ in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge ~~and exhaust penetrations [and the containment personnel air locks]. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System.~~ The 2

supply and exhaust isolation valves capable of being closed by an OPERABLE CIS. 2 1

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INSERT 2

The fuel handling accident analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. Additionally, containment closure is not assumed for a fuel handling accident inside containment. The Technical Requirements Manual includes a decay time requirement that no fuel movement will commence until 72 hours after shutdown. This ensures that the fuel handling accident analysis assumptions, including an open containment, are preserved.

The requirements of LCO 3.9.6, "Refueling Water Level," in conjunction with the minimum decay time prior to irradiated fuel movement, ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are a small fraction of the guideline values specified in 10 CFR 50.67. The acceptance limits for offsite radiation exposure are contained in Table 1 of Standard Review Plan Section 15.0.1, Rev. 0 (Ref. 2), which are a small fraction of the 10 CFR 50.67 values.

BASES

LCO (continued)

OPERABILITY requirements for this LCO ensure that the automatic ~~purge and exhaust~~ valve closure times specified in the FSAR can be achieved ~~and therefore meet the assumptions used in the safety analysis to ensure releases through the valves are terminated, such that the radiological doses are within the acceptance limit.~~ The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

~~The containment personnel air lock doors may be open during movement of [recently] irradiated fuel in the containment provided that one door is capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, one personnel airlock door will be closed following an evacuations of containment.~~

APPLICABILITY

The containment penetration requirements are applicable during movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment ~~because this is when there is a potential for the limiting~~ fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1~~, "Containment."~~ In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. ~~[~~Additionally, due to radioactive decay, ~~a fuel handling~~ accident ~~involving~~ handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~[X] days~~) will result in doses that are well within the guideline values specified in 10 CFR ~~100~~ even without containment closure capability.~~]~~ Therefore, under these conditions no requirements are placed on containment penetration status.

~~REVIEWER'S NOTE~~

~~The addition of the term "recently" associated with handling irradiated fuel in all of the containment function Technical Specification requirements is only applicable to those licensees who have demonstrated by analysis that after sufficient radioactive decay has occurred, off-site doses resulting from a fuel handling accident remain below the Standard Review Plan limits (well within 10 CFR 100).~~

BASES

APPLICABILITY (continued)

~~Additionally, licensees adding the term "recently" must make the following commitment which is consistent with NUMARC 93-01, Revision [4F], Section 11.3.6.5 "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions," subheading "Containment - Primary (PWR)/Secondary (BWR)."~~

~~"The following guidelines are included in the assessment of systems removed from service during movement of irradiated fuel:~~

~~- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.~~

~~- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure.~~

~~The purpose of the "prompt methods" mentioned above are to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."~~

3

ACTIONS

A.1~~ and A.2~~

1

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the ~~Containment Purge and Exhaust Isolation System~~ not capable of automatic actuation when ~~the purge and exhaust valves are open~~, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending movement of ~~[~~recently~]~~ irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

CIS

a required containment isolation valve in one or more penetrations with direct access from the containment atmosphere to the outside atmosphere is open

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2

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open ~~purge and exhaust~~ valves will demonstrate that the valves are not blocked from closing. Also, the Surveillance will demonstrate that each valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE automatic ~~containment purge and exhaust isolation signa~~l.

containment isolation

in penetrations with direct access from the containment atmosphere to the outside atmosphere

CIS

1

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~~[The Surveillance is performed every 7 days during movement of [recently] irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident [involving handling recently irradiated fuel] that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment in excess of those recommended by Standard Review Plan Section 15.7.4 (Ref. 3).~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

3

SR 3.9.3.2

This Surveillance demonstrates that each containment ~~purge and exhaust~~ valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. ~~[The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.4 [(Digital) or 3.3.3 (Analog)], "Miscellaneous Actuations," the Containment Purge Isolation Signal System requires a CHANNEL CHECK every 7 days and a CHANNEL FUNCTIONAL TEST every 31 days to ensure the channel OPERABILITY during refueling~~

required

isolation

in penetrations with direct access from the containment atmosphere to the outside atmosphere

CIS

1

2

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BASES

SURVEILLANCE REQUIREMENTS (continued)

operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the INSERVICE TESTING PROGRAM requirements. 2

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement. 2
----------]

These surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident [involving handling recently irradiated fuel] to limit a release of fission product radioactivity from the containment. 2

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988. 1

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1. → 2. FSAR, Section [ ] ← 15.4.3 1

2. → 3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981. ← 15.0.1, Rev. 0, July 2000 1

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment, a release of fission product radioactivity within the containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required. 2

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR ~~100~~ 50.67. Additionally, the containment structure provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions. 1

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. 2

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of ~~[~~recently~~]~~ irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must ~~always~~ remain ~~[capable of being]~~ closed. 2 2

BASES

BACKGROUND (continued)

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

INSERT 1

~~The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 42 inch purge penetration and a 42 inch exhaust penetration. The second subsystem, a minipurge system, includes an 8 inch purge penetration and an 8 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and exhaust penetrations are secured in the closed position. The two valves in each of the two minipurge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.~~

1

~~In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 42 inch purge system is used for this purpose and all valves are closed by the ESFAS in accordance with LCO 3.3.2, "Reactor Protective System (RPS) - Shutdown."~~

~~[The minipurge system remains operational in MODE 6 and all four valves are also closed by the ESFAS.~~

~~[or]~~

~~The minipurge system is not used in MODE 6. All four [8] inch valves are secured in the closed position.]~~

The ~~other~~ containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated ~~on at least one side~~. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during ~~[~~recently~~]~~ irradiated fuel movements ~~(Ref. 1)~~.

1

2

1

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident ~~[involving handling recently irradiated fuel]~~. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. ~~2~~). Fuel handling accidents, ~~analyzed in Ref. 3, include~~ dropping a

2

1

1

involve

1

1

INSERT 1

The Containment Purge and Exhaust System includes 48 inch purge supply and exhaust penetrations and 8 inch purge supply and exhaust penetrations. The second subsystem, a mini purge system, includes an 8 inch purge supply penetration and an 8 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two 48 inch valves in the purge supply and exhaust penetrations are secured in the closed position. The two 8 inch valves can be opened intermittently. The purge supply and exhaust valves (48 inch and 8 inch) are closed automatically by the containment isolation actuation signal (CIAS).

BASES

APPLICABLE SAFETY ANALYSES (continued)

INSERT 2

single irradiated fuel assembly ~~and handling tool or a heavy object onto other irradiated fuel assemblies.~~ ~~The requirements of LCO 3.9.6, "Refueling Water Level," in conjunction with minimum decay time of [72] hours prior to [irradiated fuel movement with containment closure capability or a minimum decay time of [x] days without containment closure capability], ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. The acceptance limits for offsite radiation exposure are contained in Standard Review Plan Section 15.7.4, Rev. 1 (Ref. 3), which defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values.~~

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

~~----------REVIEWER'S NOTE----------~~
~~The allowance to have containment personnel air lock doors open and penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement is based on (1) confirmatory dose calculations of a fuel handling accident as approved by the NRC staff which indicate acceptable radiological consequences and (2) commitments from the licensee to implement acceptable administrative procedures that ensure in the event of a refueling accident (even though the containment fission product control function is not required to meet acceptable dose consequences) that the open air lock can and will be promptly closed following containment evacuation and that the open penetration(s) can and will be promptly closed. The time to close penetrations or combination of penetrations shall be included in the confirmatory dose calculations.~~
~~----------~~

This LCO limits the consequences of a fuel handling accident ~~[~~involving handling recently irradiated fuel~~]~~ in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge ~~and exhaust penetrations [and the containment personnel air locks]. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System.~~ The

supply and exhaust isolation valves capable of being closed by an OPERABLE CIS.

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INSERT 2

The fuel handling accident analysis assumes movement of an irradiated fuel assembly that has not occupied part of a critical reactor core within the previous 72 hours. Additionally, containment closure is not assumed for a fuel handling accident inside containment. The Technical Requirements Manual includes a decay time requirement that no fuel movement will commence until 72 hours after shutdown. This ensures that the fuel handling accident analysis assumptions, including an open containment, are preserved.

The requirements of LCO 3.9.6, "Refueling Water Level," in conjunction with the minimum decay time prior to irradiated fuel movement, ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are a small fraction of the guideline values specified in 10 CFR 50.67. The acceptance limits for offsite radiation exposure are contained in Table 1 of Standard Review Plan Section 15.0.1, Rev. 0 (Ref. 2), which are a small fraction of the 10 CFR 50.67 values.

BASES

LCO (continued)

OPERABILITY requirements for this LCO ensure that the automatic ~~purge and exhaust~~ valve closure times specified in the FSAR can be achieved ~~and therefore meet the assumptions used in the safety analysis to ensure releases through the valves are terminated, such that the radiological doses are within the acceptance limit.~~ The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

~~The containment personnel air lock doors may be open during movement of [recently] irradiated fuel in the containment provided that one door is capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, one personnel airlock door will be closed following an evacuations of containment.~~

APPLICABILITY

The containment penetration requirements are applicable during movement of ~~[~~recently~]~~ irradiated fuel assemblies within containment ~~because this is when there is a potential for the limiting~~ fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1~~, "Containment."~~ In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. ~~[~~Additionally, due to radioactive decay, ~~a fuel handling~~ accident ~~involving~~ handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~[X] days~~) will result in doses that are well within the guideline values specified in 10 CFR ~~100~~ even without containment closure capability.~~]~~ Therefore, under these conditions no requirements are placed on containment penetration status.

~~REVIEWER'S NOTE~~

~~The addition of the term "recently" associated with handling irradiated fuel in all of the containment function Technical Specification requirements is only applicable to those licensees who have demonstrated by analysis that after sufficient radioactive decay has occurred, off-site doses resulting from a fuel handling accident remain below the Standard Review Plan limits (well within 10 CFR 100).~~

BASES

APPLICABILITY (continued)

~~Additionally, licensees adding the term "recently" must make the following commitment which is consistent with NUMARC 93-01, Revision [4F], Section 11.3.6.5 "Safety Assessment for Removal of Equipment from Service During Shutdown Conditions," subheading "Containment - Primary (PWR)/Secondary (BWR)."~~

~~"The following guidelines are included in the assessment of systems removed from service during movement of irradiated fuel:~~

~~- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.~~

~~- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure.~~

~~The purpose of the "prompt methods" mentioned above are to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."~~

3

ACTIONS

A.1~~ and A.2~~

1

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the ~~Containment Purge and Exhaust Isolation System~~ not capable of automatic actuation when ~~the purge and exhaust valves are open~~, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending movement of ~~[~~recently~]~~ irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

CIS

a required containment isolation valve in one or more penetrations with direct access from the containment atmosphere to the outside atmosphere is open

1

2

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open ~~purge and exhaust~~ valves will demonstrate that the valves are not blocked from closing. Also, the Surveillance will demonstrate that each valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE automatic ~~containment purge and exhaust isolation signa~~l.

containment isolation

in penetrations with direct access from the containment atmosphere to the outside atmosphere

CIS

1

1

~~[The Surveillance is performed every 7 days during movement of [recently] irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident [involving handling recently irradiated fuel] that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment in excess of those recommended by Standard Review Plan Section 15.7.4 (Ref. 3).~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~

3

SR 3.9.3.2

This Surveillance demonstrates that each containment ~~purge and exhaust~~ valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. ~~[The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.4 [(Digital) or 3.3.3 (Analog)], "Miscellaneous Actuations," the Containment Purge Isolation Signal System requires a CHANNEL CHECK every 7 days and a CHANNEL FUNCTIONAL TEST every 31 days to ensure the channel OPERABILITY during refueling~~

required

isolation

in penetrations with direct access from the containment atmosphere to the outside atmosphere

CIS

1

2

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the INSERVICE TESTING PROGRAM requirements. 2

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------- REVIEWER'S NOTE ----------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement. 2
----------]

These surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident [involving handling recently irradiated fuel] to limit a release of fission product radioactivity from the containment. 2

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988. 1

U
1. 2. FSAR, Section [ ] . ← 15.7.4 1

2. 3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981. ← 15.0.1, Rev. 0, July 2000 1

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.3, BASES, CONTAINMENT PENETRATIONS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.3, CONTAINMENT PENETRATIONS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 4

3.9.4, Shutdown Cooling (SDC) and Coolant Circulation – High Water Level

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REFUELING OPERATIONS

SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

SDC

LCO 3.9.4 ~~3.9.8.1~~ ~~At least~~ one ~~shutdown cooling~~ loop shall be OPERABLE and in operation*.

with

Applicability **APPLICABILITY:** MODE 6 ~~when~~ the water level above the top of ~~irradiated fuel assemblies seated within~~ the reactor ~~pressure~~ vessel flange ~~is greater than or equal to~~ 23 feet.

M01

< ≥

Immediately — M04

Add ACTION A.1 — M02

Add ACTION A.3 — A02

A02

ACTION:

required SDC inoperable or not

ACTION A ~~a.~~ ~~With less than~~ one ~~shutdown cooling~~ loop in operation, suspend ~~all operations involving an increase in reactor decay heat load or~~ operations that would cause introduction into the RCS, ~~coolant~~ with boron concentration less than required to meet the boron concentration of ~~Technical Specification~~ 3.9.1.

of coolant

ACTION A.2

LCO

ACTION A.6 Close ~~all containment~~ penetration~~s~~ providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

each

L01

with a manual or automatic isolation valve, blind flange, or equivalent — Add ACTIONS A.4, A.5, A.6

~~b.~~ ~~The provisions of Specification 3.0.3 are not applicable.~~ A03

SURVEILLANCE REQUIREMENTS

Verify SDC

SR 3.9.4.1 ~~4.9.8.1~~ ~~At least~~ one ~~shutdown cooling loop shall be verified to be~~ in operation and circulating reactor coolant at a flow rate ~~of greater than or equal to~~ 3000 gpm in accordance with the Surveillance Frequency Control Program.

≥

SDC

SR 3.9.4.2 ~~4.9.8.1.1~~ Verify required ~~shutdown cooling~~ loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

required SDC ≤

SR 3.9.4 Note * The ~~shutdown cooling~~ loop may be removed from operation for ~~up to~~ 1 hour per 8 hour period ~~during the performance of CORE ALTERATIONS in the vicinity of reactor pressure vessel hot legs,~~ provided no operations are permitted that would cause introduction into the ~~RCS, coolant~~ with boron concentration less than required to meet the ~~SHUTDOWN MARGIN~~ of ~~Technical Specification~~ 3.9.1.

L02

of coolant

that

minimum required boron concentration

LCO

Reactor Coolant System

A01

REFUELING OPERATIONS

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

LCO 3.9.4 ~~3.9.8.1~~ ~~At least~~ one ~~shutdown cooling~~ loop shall be OPERABLE and in operation.*

Applicability **APPLICABILITY:** MODE 6 ~~when~~ the water level above the top of the reactor ~~pressure~~ vessel flange ~~is greater than or equal to~~ 23 feet.

ACTION:

ACTION A ~~With no shutdown cooling~~ loop ~~OPERABLE~~ and in operation, suspend ~~all operations involving an increase in reactor decay heat load or~~ operations that would cause introduction into the RCS, ~~coolant~~ with boron concentration less than required to meet the boron concentration of ~~Technical Specification~~ 3.9.1 ~~and within 1 hour~~ initiate corrective action to return the required shutdown cooling loop to OPERABLE and operating status ~~as soon as possible~~. Close ~~all containment~~ penetration~~s~~ providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1, SR 3.9.5.2 ~~4.9.8.1~~ In accordance with the Surveillance Frequency Control Program:

SR 3.9.5.1 ~~a.~~ ~~At least~~ one ~~shutdown cooling~~ loop ~~shall be verified to be~~ in operation

SR 3.9.5.1 ~~b.~~ ~~The total~~ flow rate of reactor coolant ~~to the reactor pressure vessel shall be verified to be greater than or equal to~~ 3000 gpm.**

SR 3.9.5.2 ~~c.~~ Verify required ~~shutdown cooling~~ loop locations susceptible to gas accumulation are sufficiently filled with water.

LCO 3.9.4 Note * The ~~shutdown cooling~~ loop may be removed from operation for ~~up to~~ 1 hour per 8-hour period ~~during the performance of CORE ALTERATIONS in the vicinity of reactor pressure vessel hot legs,~~ provided no operations are permitted that would cause introduction into the ~~RCS, coolant~~ with boron concentration less than required to meet the ~~SHUTDOWN MARGIN~~ of ~~Technical Specification~~ 3.9.1.

SR 3.9.4.1 Note ** ~~The~~ reactor coolant flow rate ~~requirement~~ may be reduced to 1850 gpm ~~if the following conditions are satisfied before the reduced requirement is implemented:~~ the reactor ~~has been determined to have been~~ subcritical ~~for at least~~ 125 hours, the maximum RCS temperature is ≤ 117°F, and ~~the~~ temperature ~~of CCW~~ to the ~~shutdown cooling~~ heat exchanger is ≤ 87°F.

DISCUSSION OF CHANGES
ITS 3.9.4, SHUTDOWN COOLING (SDC) AND COOLANT CIRCULATION – HIGH WATER LEVEL

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 Unit 1 CTS 3.9.8.1 Action a. states, in part, that with less than one shutdown cooling (SDC) loop in operation, suspend all operations involving an increase in the reactor decay heat load of the RCS. Similarly, Unit 2 CTS 3.9.8.1 Action states, in part, that with no SDC loop in operation, suspend all operations involving an increase in the reactor decay heat load of the RCS. ITS 3.9.4 Required Action A.3 states, in part, that with one required SDC loop inoperable or not in operation, suspend loading irradiated fuel assemblies in the core. This changes the CTS by requiring that the loading of irradiated fuel assemblies be suspended instead of requiring that all operations involving an increase in the reactor decay heat load be suspended.

This change is acceptable because the requirements have not changed. The reactor decay heat load is generated only by irradiated fuel. The only method of increasing the decay heat load of a reactor in MODE 6 is to load additional irradiated fuel assemblies into the core. Therefore, the CTS and ITS requirements are equivalent. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 **Unit 1 only:** CTS 3.9.8.1 Action b. states "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.4 does not include this statement. This changes CTS by deleting the Specification 3.0.3 exception. This change is acceptable because the technical requirements have not changed. ITS LCO 3.0.3 is not applicable in MODE 6. Therefore, the CTS LCO 3.0.3 exception is not needed. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS 3.9.8.1 requires at least one SDC loop to be OPERABLE and in operation in MODE 6 when the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is ≥ 23 feet. ITS 3.9.4 requires one SDC loop to be OPERABLE and in operation when water level is ≥ 23 feet above the top of the reactor vessel flange. This changes the CTS by changing the point at which one SDC loop is required to be OPERABLE and in operation.

The purpose of CTS 3.9.8.1 is to ensure adequate SDC is available and in operation for heat removal and coolant circulation. CTS 3.9.8.1 and CTS 3.9.8.2

provide the requirements when water level is ≥ 23 feet and < 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel, respectively. When water level is ≥ 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel, only one SDC loop is required to be OPERABLE and in operation. When water level is < 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel, two SDC loops are required to be OPERABLE, and one must be in operation. In ITS 3.9.4 and ITS 3.9.5, the equivalent ITS requirements, the water level reference point is the top of the reactor vessel flange, not the top of the irradiated fuel assemblies seated within the reactor pressure vessel. Changing this reference point effectively requires a larger complement of SDC loops to be OPERABLE between 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel and 23 feet above the top of the reactor vessel flange. Therefore, this change is acceptable because more loops will be required to be OPERABLE under certain water level conditions to ensure the decay heat can be removed and the coolant circulated. This change is designated more restrictive because more SDC loops are required OPERABLE in the ITS under certain water level conditions than were required in the CTS.

M02 **Unit 1 only:** CTS 3.9.8.1 Actions do not include an action to immediately initiate action to satisfy the SDC loop requirements in the event the SDC loop requirements are not met. ITS 3.9.4 Required Action A.1 requires that action be immediately initiated to satisfy the SDC loop requirements. This changes the CTS by requiring action be taken immediately to satisfy the SDC loop requirements.

The purpose of CTS 3.9.8.1 is to ensure that adequate decay heat removal and coolant circulation are available in MODE 6. Although decay heat is removed from the Reactor Coolant System via natural circulation to the bulk of water contained in the refueling cavity, this method of heat transfer can continue for only a discrete amount of time before boiling would occur. In addition, with no forced circulation, RCS flow may not be adequate to support coolant mixing to maintain uniform boron concentrations. This change is acceptable because it requires that action be initiated to restore the SDC loop requirements in order to restore forced coolant flow and heat removal. This change is designated as more restrictive because additional actions will be required in the ITS than are required in the CTS.

M03 **Unit 2 only**: CTS 3.9.8.1 Action states, in part, within one (1) hour initiate corrective action to return the required shutdown cooling loop to operation. ITS 3.9.4 Required Action A.1 states immediately initiate action to restore one SDC loop to OPERABLE status and operation. This changes the CTS by revising the Actions to “immediately” require actions to be taken when a required SDC loop is not in operation. | R1 | R1

This change is acceptable because it provides appropriate immediate actions for a required SDC loop not in operation. This action is required to assure continued safe operation. Action to restore one SDC loop to OPERABLE status and operation is necessary to be able to remove the decay heat generated by the reactor. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal. This change is designated as more restrictive

because the time to complete the actions is added in the ITS that is not required in the CTS.

M04 CTS 3.9.8.1 Actions do not include a Completion Time to immediately initiate action to suspend operations that would cause the introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1. ITS 3.9.4 Required Action A.2 requires that action be immediately initiated to suspend operations that would cause introduction of coolant into the RCS that does not meet the minimum boron concentration requirement. This changes the CTS by requiring that action be taken immediately to perform a Required Action.

The purpose of CTS 3.9.8.1 is to ensure that adequate decay heat removal and coolant circulation are available in MODE 6. If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Immediately suspending operations that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. This change is designated as more restrictive because a Completion Time in ITS is being added to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 4 – Relaxation of Required Action)* Unit 1 CTS 3.9.8.1 Action a. states, in part, that with less than one SDC loop in operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. Similarly, Unit 2 CTS 3.9.8.1 Action states, in part, that with no SDC loop in operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. ITS 3.9.4 Required Actions A.4, A.5, and A.6 state that with the SDC loop requirements not met, within 4 hours close and secure the equipment hatch with at least four bolts, close one door in each air lock, and verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by an OPERABLE Containment Isolation System. This changes the CTS Actions by specifying the applicable containment penetration requirements to satisfy “each penetration” and by separating the CTS action into allowing penetrations capable of being closed by an OPERABLE Containment Isolation System to remain open when the SDC requirements are not met.

The purpose of the CTS 3.9.8.1 Action is to ensure that radioactive material does not escape the containment should the SDC requirements continue to not be met and boiling occurs in the core. Therefore, containment penetrations are closed to seal the containment. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of an accident occurring during the repair period. The Required Actions are consistent with the actions taken for containment closure in CTS 3.9.4 (ITS 3.9.3). Penetrations which can be closed by an OPERABLE Containment Isolation System do not need to be closed if SDC is inoperable, since the presence of radioactivity in the containment will cause the valves to close automatically, thus performing the isolation function. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L02 *(Category 3 – Relaxation of Completion Time)* CTS 3.9.8.1 Footnote * states, in part, that the SDC loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs, provided no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.9.1. ITS LCO 3.9.4 Note states that the required SDC loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1, "Boron Concentration." The allowance to remove SDC from operation is no longer restricted to CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs. This changes the CTS by allowing the SDC loop to be removed from operation during conditions other than CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs. | R1

The purpose of the allowance to remove the SDC loop from operation for up to 1 hour per 8 hour period is to perform certain activities that cannot be performed with a SDC loop in service; for example, core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. This change is acceptable because, during the 1 hour period, decay heat is removed by natural convection due to the large mass of water in the refueling cavity and appropriate controls are continue to be provided in the Technical Specifications during periods when SDC is not in operation prohibiting activities that would cause introduction of coolant into the RCS with boron concentration less than the minimum required boron concentration to preserve SHUTDOWN MARGIN and avoid unexpected reactivity changes. This change is designated as less restrictive because the scope of the current allowance is expanded to be less stringent in the ITS than applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.9 REFUELING OPERATIONS

3.9.4 Shutdown Cooling (SDC) and Coolant Circulation - High Water Level

3.9.8.1 LCO 3.9.4 One SDC loop shall be OPERABLE and in operation.

(RCS) 3

----------NOTE----------

3.9.8.1 Footnote * The required SDC loop may be removed from operation for ≤ 1 hour per ~~[~~8~~]~~ hour period, provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1~~.~~ , "Boron Concentration." 2 3

Applicability APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.8.1 Action DOC M01 A. One required SDC loop inoperable or not in operation.	A.1 Initiate action to restore SDC loop to OPERABLE status and operation.	Immediately
	AND	
3.9.8.1 Action a. DOC M04	A.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
DOC M02	A.3 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01	A.4 Close equipment hatch and secure with ~~[~~four~~]~~ bolts.	4 hours (2)
	AND	
DOC L01	A.5 Close one door in each air lock.	4 hours
	AND	
3.9.8.1 Action	A.6~~.1~~ ~~Close~~ each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent~~.~~ ; or	4 hours (4)
	~~OR~~	
DOC L01	~~A.6.2 Verify each penetration~~ is capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Isolation System.	~~4 hours~~ (4) (1)

Verify

is either closed

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.9.8.1	SR 3.9.4.1	Verify one SDC loop is in operation and circulating reactor coolant at a flow rate of ≥ ~~[2200]~~ 3000 gpm.	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2
4.9.8.1.1	SR 3.9.4.2	Verify required SDC loop locations susceptible to gas accumulation are sufficiently filled with water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	2 2

3.9 REFUELING OPERATIONS

3.9.4 Shutdown Cooling (SDC) and Coolant Circulation - High Water Level

3.9.8.1 LCO 3.9.4 One SDC loop shall be OPERABLE and in operation.

----------NOTE----------

3.9.8.1 Footnote * The required SDC loop may be removed from operation for ≤ 1 hour per ~~[~~8~~]~~ hour period, provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1~~.~~ , "Boron Concentration."

Applicability APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.8.1 Action DOC L03 A. One required SDC loop inoperable or not in operation.	A.1 Initiate action to restore SDC loop to OPERABLE status and operation.	Immediately
	AND	
3.9.8.1 Action DOC M04	A.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	A.3 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC L01	A.4 Close equipment hatch and secure with ~~[~~four~~]~~ bolts.	4 hours (2)
	AND	
DOC L01	A.5 Close one door in each air lock.	4 hours
	AND	
3.9.8.1 Action	A.6~~.1~~ ~~Close~~ each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent~~.~~	4 hours (4)
	~~OR~~	
DOC L01	~~A.6.2 Verify each penetration~~ is capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Isolation System.	~~4 hours~~ (4) (1)

Verify

is either closed

; or

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one SDC loop is in operation and circulating reactor coolant at a flow rate of ≥ ~~[2200]~~ 3000 gpm. ----------NOTE---------- Reactor coolant flow rate may be reduced to ≥ 1850 gpm when the reactor is subcritical ≥ 125 hours, the maximum RCS temperature is ≤ 117°F, and component cooling water temperature to the SDC heat exchanger is ≤ 87°F. ----------	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
SR 3.9.4.2	Verify required SDC loop locations susceptible to gas accumulation are sufficiently filled with water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

4.9.8.1.a
4.9.8.1.b
4.9.8.1.b Footnote **
4.9.8.1.c
2
5
2
2
2

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.4, SHUTDOWN COOLING (SDC) AND COOLANT CIRCULATION – HIGH WATER LEVEL

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Change made to be consistent with the ISTS format as specified in the Writer's Guide for the Improved Standard Technical Specifications, TSTF GG 05 01 and other Specifications; for example, LCO 3.8.2.a reference to LCO 3.8.10, “Distribution Systems – Shutdown.”

4. ISTS 3.9.4 Required Actions A.6.1 and A.6.2 are connected by an "OR" logical connector, such that either one can be performed to meet the requirements of the ACTION. However, the two Required Actions as presented in the ISTS are applicable to all the applicable containment penetrations; either Required Action A.6.1 or Required Action A.6.2 must be performed. Thus, the ISTS presentation will not allow one penetration to be isolated by use of a manual valve and another penetration to be capable of being closed by an OPERABLE Containment Isolation System. This is not the intent of the requirement. The requirement is based on ISTS LCO 3.9.3, which requires each penetration to be either: a) closed by a manual or automatic isolation valve, blind flange, or equivalent; or b) capable of being closed by an OPERABLE Containment Isolation System. For consistency with the actual LCO requirement, ISTS 3.9.4 Required Actions A.6.1 and A.6.2 have been combined into a single Required Action in ITS 3.9.4 Required Action A.6. This will allow each of the applicable containment penetrations to be either isolated or capable of being isolated, as required, to ensure containment closure can be achieved.

5. **Unit 2 only:** A Note is added to ITS SR 3.9.4.1 that allows a reduction of reactor coolant flow rate when specific conditions are met consistent with the St. Lucie Plant Unit 2 current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.9 REFUELING OPERATIONS

B 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation - High Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS) ~~, as required by GDC 34,~~ to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water (CCW) System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System, as well as adjustments in CCW temperature and flow.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to a resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One ~~train~~ loop of the SDC System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit the SDC pump to be removed from operation for short durations under the condition that the boron concentration is not diluted. This conditional stopping of the SDC pump does not result in a challenge to the fission product barrier.

SDC and Coolant Circulation - High Water Level satisfies Criterion ~~2~~ 4 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

Only one SDC loop is required for decay heat removal in MODE 6, with water level ≥ 23 ft above the top of the reactor vessel flange. Only one SDC loop is required because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one SDC loop must be OPERABLE and in operation to provide:

a. Removal of decay heat, and

b. Mixing of borated coolant to minimize the possibility of a criticality~~, and~~

~~c. Indication of reactor coolant temperature~~.

An OPERABLE SDC loop includes an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path ~~and to determine the low end temperature~~. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. Management of gas voids is important to SDC System OPER~~E~~ABILITY.

Additionally, since the SDC System is a manually operated system (i.e., it is not automatically actuated), each SDC loop is OPERABLE if it can be manually aligned (remote or local) to the SDC mode.

Both SDC pumps may be aligned to the Refueling Water Storage Tank to support filing or draining the refueling cavity or for performance of required testing.

The LCO is modified by a Note that allows the required operating SDC loop to be removed from operation for up to 1 hour in each 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1~~.~~ Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One SDC loop must be in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Water Level." Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS)~~, and Section 3.5, Emergency Core Cooling Systems (ECCS)~~. SDC loop requirements in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, are located in LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

BASES

ACTIONS

SDC loop requirements are met by having one SDC loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If one required SDC loop is inoperable or not in operation, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and to operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

A.2

If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.3

If SDC loop requirements are not met, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase the decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.4, A.5~~, A.6.1,~~ and A.6~~.2~~ 6

If no SDC loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with ~~[~~four~~]~~ bolts, 2

b. One door in each air lock must be closed, and

BASES

ACTIONS (continued)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Isolation System. 1

With SDC loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most SDC problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the SDC loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. ~~[The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the SDC System.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~----------]~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.4.2

SDC System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the required SDC loop(s) and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

Selection of SDC System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The SDC System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the SDC System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

SDC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible

BASES

SURVEILLANCE REQUIREMENTS (continued)

location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the SDC System piping and the procedural controls governing system operation.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 2

REFERENCES 1. FSAR, Section ~~[~~ ~~]~~. 1 2

U 15.2.4

B 3.9 REFUELING OPERATIONS

B 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation - High Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS)~~, as required by GDC 34,~~ to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System.

(CCW)

, as well as adjustments in CCW temperature and flow

1

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to a resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One ~~train~~ of the SDC System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit the SDC pump to be removed from operation for short durations under the condition that the boron concentration is not diluted. This conditional stopping of the SDC pump does not result in a challenge to the fission product barrier.

loop

1

SDC and Coolant Circulation - High Water Level satisfies Criterion ~~2~~ 4 of 10 CFR 50.36(c)(2)(ii).

3

1

BASES

LCO

Only one SDC loop is required for decay heat removal in MODE 6, with water level ≥ 23 ft above the top of the reactor vessel flange. Only one SDC loop is required because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one SDC loop must be OPERABLE and in operation to provide:

a. Removal of decay heat, and

b. Mixing of borated coolant to minimize the possibility of a criticality~~, and~~

~~c. Indication of reactor coolant temperature~~.

An OPERABLE SDC loop includes an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path ~~and to determine the low end temperature~~. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. Management of gas voids is important to SDC System OPER~~E~~ABILITY.

at least one of

Additionally, since the SDC System is a manually operated system (i.e., it is not automatically actuated), each SDC loop is OPERABLE if it can be manually aligned (remote or local) to the SDC mode.

Both SDC pumps may be aligned to the Refueling Water Storage Tank to support filing or draining the refueling cavity or for performance of required testing.

The LCO is modified by a Note that allows the required operating SDC loop to be removed from operation for up to 1 hour in each 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1~~.~~ Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

, Boron Concentration."

APPLICABILITY

One SDC loop must be in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Water Level." Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS)~~, and Section 3.5, Emergency Core Cooling Systems (ECCS)~~. SDC loop requirements in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, are located in LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

BASES

ACTIONS

SDC loop requirements are met by having one SDC loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If one required SDC loop is inoperable or not in operation, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and to operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

A.2

If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.3

If SDC loop requirements are not met, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase the decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.4, A.5~~, A.6.1,~~ and A.6~~.2~~ 6

If no SDC loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with ~~[~~four~~]~~ bolts, 2

b. One door in each air lock must be closed, and

BASES

ACTIONS (continued)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Isolation System. 1

With SDC loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most SDC problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the SDC loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. ~~[The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the SDC System.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

The SR is modified by a Note that provides an allowance to reduce the required RCS flow rate when specific conditions are met. The time after reactor shutdown, RCS temperature, and CCW temperature to the shutdown cooling heat exchanger are initial conditions to ensure a reduced SDC flow rate will not result in a temperature transient exceeding the established refueling operational temperature limit. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.4.2

SDC System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the required SDC loop(s) and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

Selection of SDC System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

The SDC System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the SDC System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

SDC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible

BASES

SURVEILLANCE REQUIREMENTS (continued)

location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the SDC System piping and the procedural controls governing system operation.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~ 2

REFERENCES 1. FSAR, Section ~~[~~ ~~]~~. U 15.4.6 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.4, BASES, SHUTDOWN COOLING (SDC) AND COOLANT CIRCULATION – HIGH WATER LEVEL

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The SDC and Coolant Circulation Technical Specification requirements in MODE 6 do not represent a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, these requirements do not meet Criterion 2 of 10 CFR 50.35(c)(2)(ii) at St. Lucie Plant Unit 1 and Unit 2. The SDC and Coolant Circulation Technical Specification requirements in MODE 6 do meet Criterion 4 of 10 CFR 50.35(c)(2)(ii) because this system is an important contributor to risk reduction in the plant specific probabilistic risk assessment. Therefore, the ISTS reference to the 10 CFR 50.36(c)(2)(ii) criterion is changed from 2 to 4 in the ITS.

4. Typographical/grammatical error corrected.

5. Change made to be consistent with the ISTS format as specified in the Writer's Guide for the Improved Standard Technical Specifications, TSTF GG 05 01 and other ISTS Bases (e.g., ISTS Bases 3.9.1 refers to LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level").

6. Changes are made to reflect changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.4, SHUTDOWN COOLING (SDC) AND COOLANT CIRCULATION – HIGH WATER LEVEL

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 5

3.9.5, Shutdown Cooling (SDC) and Coolant Circulation – Low Water Level

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two ~~independent~~ ~~shutdown cooling~~ loops shall be OPERABLE and ~~at least~~ one ~~shutdown cooling~~ loop shall be in operation.*

SDC | with | SDC | Add LCO 3.9.5 note 1 | L02

APPLICABILITY: MODE 6 ~~when~~ the water level above the top of ~~irradiated fuel assemblies seated within the~~ reactor ~~pressure~~ vessel ~~is less than~~ 23 feet.

flange | < | M01

ACTION:

One | SDC | inoperable | immediately | M02

ACTION A
ACTION A.1

a. ~~With less than the required~~ ~~shutdown cooling~~ loop~~s~~ ~~OPERABLE~~, ~~within one (1) hour 1)~~ initiate ~~corrective~~ action to ~~return the required~~ loop~~s~~ to OPERABLE status, ~~2)~~ establish ~~greater than or equal to~~ 23 feet of water above ~~irradiated fuel assemblies seated within~~ the reactor ~~pressure~~ vessel.

restore SDC | ACTION A.2 | ≥ | flange | top of

SDC | OPERABLE or | immediately | M02

ACTION B
ACTION B.1

b. ~~With~~ no ~~shutdown cooling~~ loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of ~~Technical Specification~~ 3.9.1. and ~~within one (1) hour~~ initiate ~~corrective~~ action to ~~return the required shutdown cooling~~ loop to operation. ~~Close all containment~~ penetration~~s~~ providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

of | LCO | ACTION B.2 | ACTION B.5 | Verify each | restore SDC | OPERABLE status and to | is either closed with a manual or automatic isolation valve, blind flange, or equivalent | Add ACTIONS B.3, B.4 | L01

c. ~~The provisions of Specification 3.0.3 are not applicable.~~

Immediately | A02 | M02

SURVEILLANCE REQUIREMENTS

Verify | SDC

SR 3.9.5.1 ~~4.9.8.2~~ ~~At least~~ one ~~shutdown cooling~~ loop ~~shall be verified to be~~ in operation and circulating reactor coolant at a flow rate ~~of greater than or equal to~~ 3000 gpm in accordance with the Surveillance Frequency Control Program.

≥

SDC

SR 3.9.5.3 ~~4.9.8.2.1~~ Verify ~~shutdown cooling~~ loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

Add SR 3.9.5.2 | M03

SDC

LCO 3.9.5 Note 2 * One required ~~shutdown cooling~~ loop may be inoperable for up to 2 hours for surveillance testing, provided that the other shutdown cooling loop is OPERABLE and in operation.

A01

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two ~~independent~~ ~~shutdown~~ cooling loops shall be OPERABLE and ~~at least~~ one ~~shutdown~~ cooling loop shall be in operation.**

Add LCO 3.9.5 note 1 — L02

APPLICABILITY: MODE 6 ~~when~~ the water level above the top of ~~the~~ reactor ~~pressure~~ vessel flange ~~is less than~~ 23 feet.

ACTION:

ACTION A
ACTION A.1
ACTION A.2

a. ~~With less than the required~~ ~~shutdown~~ cooling loop~~s~~ OPERABLE, ~~within 1 hour~~ initiate ~~corrective~~ action to ~~return the required~~ loop~~s~~ to OPERABLE status, or ~~to~~ establish ~~greater than or equal to~~ 23 feet of water above the reactor ~~pressure~~ vessel flange~~, as soon as possible~~. — M02

ACTION B
ACTION B.1
ACTION B.2
ACTION B.5

b. ~~With~~ no ~~shutdown~~ cooling loop in operation, suspend operations that would cause introduction into the RCS~~,~~ coolant with boron concentration less than required to meet the boron concentration of ~~Technical Specification~~ 3.9.1 and ~~within 1 hour~~ initiate ~~corrective~~ action to ~~return the required shutdown cooling~~ loop to operation. ~~Close all containment~~ penetration~~s~~ providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. — L01

Immediately — M02

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1, SR 3.9.5.3

4.9.8.2 In accordance with the Surveillance Frequency Control Program:

SR 3.9.5.1

a. ~~At least~~ one ~~shutdown~~ cooling loop ~~shall be verified to be~~ in operation~~.~~

SR 3.9.5.1

b. ~~The total~~ flow rate of reactor coolant ~~to the reactor pressure vessel shall be verified to be greater than or equal to~~ 3000 gpm.*

SR 3.9.5.3

c. Verify ~~shutdown~~ cooling trains locations susceptible to gas accumulation are sufficiently filled with water.

Add SR 3.9.5.2 — M03

SR 3.9.5.1 Note

* ~~The~~ reactor coolant flow rate ~~requirement~~ may be reduced to 1850 gpm ~~if the following conditions are satisfied before the reduced requirement is implemented:~~ the reactor ~~has been determined to have been~~ subcritical ~~for at least~~ 125 hours, the maximum RCS temperature is ≤ 117°F, and ~~the~~ temperature ~~of CCW~~ to the ~~shutdown~~ cooling heat exchanger is ≤ 87°F.

LCO 3.9.5 Note 2

** One required ~~shutdown~~ cooling loop may be inoperable for up to 2 hours for surveillance testing, provided that the other shutdown cooling loop is OPERABLE and in operation.

DISCUSSION OF CHANGES
ITS 3.9.5, SHUTDOWN COOLING (SDC) AND COOLANT CIRCULATION – LOW WATER LEVEL

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 **Unit 1 only:** CTS 3.9.8.2 Action c. states "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.5 does not include this statement. This changes CTS by deleting the Specification 3.0.3 exception. This change is acceptable because the technical requirements have not changed. ITS LCO 3.0.3 is not applicable in MODE 6. Therefore, the CTS LCO 3.0.3 exception is not needed. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS 3.9.8.2 requires two shutdown cooling (SDC) loops to be OPERABLE and one SDC loop in operation in MODE 6 when the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is < 23 feet. ITS 3.9.5 requires two SDC loops to be OPERABLE and one SDC loop in operation when water level is < 23 feet above the top of the reactor vessel flange. This changes the CTS by changing the point at which two SDC loops are required to be OPERABLE and one in operation.

The purpose of CTS 3.9.8.2 is to ensure adequate SDC is available and in operation for heat removal and coolant circulation. CTS 3.9.8.1 and CTS 3.9.8.2 provide the requirements when water level is ≥ 23 feet and < 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel, respectively. When water level is ≥ 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel, only one SDC loop is required to be OPERABLE and in operation. When water level is < 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel, two SDC loops are required to be OPERABLE, and one must be in operation. In ITS 3.9.4 and ITS 3.9.5, the equivalent ITS requirements, the water level reference point is the top of the reactor vessel flange, not the top of the irradiated fuel assemblies seated within the reactor pressure vessel. Changing this reference point effectively requires a larger complement of SDC loops to be OPERABLE between 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel and 23 feet above the top of the reactor vessel flange. Therefore, this change is acceptable because more loops will be required to be OPERABLE under certain water level conditions to ensure the decay heat can be removed and the coolant circulated. This change is designated more restrictive because more SDC loops are required OPERABLE in the ITS under certain water level conditions than were required in the CTS.

M02 CTS 3.9.8.2 Action a. states, in part, that with one SDC loop inoperable, within one (1) hour initiate corrective action to return the required loop to OPERABLE status or to restore reactor cavity water level to greater than or equal to the limit. CTS 3.9.8.2 Action b. states that with no SDC loop OPERABLE or in operation, within one (1) hour initiate corrective action to return the required shutdown cooling loop to operation. ITS 3.9.5 ACTIONS A and B require similar actions but require the initiation of the Required Actions immediately.

CTS 3.9.8.2 Action b does not include a Completion Time to initiate action to suspend operations that would cause the introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1. ITS 3.9.5 Required Action B.1 requires that action be immediately initiated to suspend operations that would cause introduction of coolant into the RCS that does not meet the minimum boron concentration requirement. This changes the CTS by requiring that actions be taken immediately to satisfy the SDC loop requirements.

The purpose of CTS 3.9.8.2 is to ensure that adequate decay heat removal and coolant circulation are available in MODE 6. If the SDC loop requirements are not met, there will be no forced circulation to provide decay heat removal or coolant circulation to maintain uniform boron concentrations. Immediately initiating action to restore SDC loops to OPERABLE status with one in operation and immediately suspending operations that could result in failure to meet the minimum boron concentration limit are required to assure continued safe operation. The immediate Completion Time reflects the importance of restoring operation for decay heat removal and suspending the specified operations. This change is designated as more restrictive because the immediate Completion Time in ITS is more restrictive than the CTS Completion Time of 1 hour.

M03 The CTS 3.9.8.2 requires two independent SDC loops to be OPERABLE. ITS SR 3.9.5.2 requires periodic verification of correct breaker alignment and that indicated power is available to the required SDC pump not in operation in accordance with the Surveillance Frequency Control Program. This changes the CTS by adding a Surveillance Requirement.

The purpose of ITS 3.9.5 is to require one SDC loop to be in operation and one SDC loop to be held in readiness should it be needed. This change is necessary to ensure that the SDC loop that is in standby will be ready should it be needed to provide decay heat removal and coolant circulation.

PSL controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program (SFCP) per CTS 6.8.4.o (Unit 1) and CTS 6.8.4.q (Unit 2). Therefore, SR 3.9.5.2 will be performed at a Frequency in accordance with the Surveillance Frequency Control Program with an initial Frequency of 7 days consistent with the ISTS SR 3.9.5.2 Frequency. The initial Frequency is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This change is designated as more restrictive because it adds a new Surveillance Requirement to the CTS.

DISCUSSION OF CHANGES
ITS 3.9.5, SHUTDOWN COOLING (SDC) AND COOLANT CIRCULATION – LOW WATER LEVEL

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* ITS 3.9.5 is modified by a Note (Note 1) that allows all SDC pumps to be removed from operation for ≤ 15 minutes when switching from one loop to another, provided several conditions are met. CTS 3.9.8.2 does not contain this allowance. This changes the CTS by allowing the LCO to not be met under certain situations.

The purpose of CTS 3.9.8.2 is to ensure sufficient decay heat removal is available in the specified MODE and condition. The ITS Note allows an operational evolution for switching from one loop to another while in the Applicability of the Specification. This evolution is necessary to demonstrate SDC OPERABILITY. This change is acceptable because the time period is short and specific controls are required to preclude reaching saturation conditions and to preclude reducing the required RCS boron concentration. The circumstances for stopping both SDC pumps are to be limited to 15 minutes when switching SDC loops and requires the core outlet temperature to be maintained > 10°F below saturation temperature. The Note also prohibits boron dilution by introduction of coolant into the RCS with boron concentration less than that required to meet the minimum boron concentration of LCO 3.9.1, Boron Concentration, and draining operations that could further reduce RCS water volume. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L02 *(Category 4 – Relaxation of Required Action)* CTS 3.9.8.2 Action b. states, in part, that with no SDC loop in operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. ITS 3.9.5 Required Actions B.3, B.4, and B.5 state that with the SDC loop requirements not met, within 4 hours close and secure the equipment hatch with at least four bolts, close one door in each air lock, and verify each penetration providing direct access from the containment atmosphere to the outside atmosphere is either closed with a manual or automatic isolation valve, blind flange, or equivalent, or is capable of being closed by an OPERABLE Containment Isolation System. This changes the CTS Actions by specifying the applicable containment penetration requirements to satisfy “each containment penetration” and by separating the CTS action into allowing penetrations capable of being closed by an OPERABLE Containment Isolation System to remain open when the SDC requirements are not met.

The purpose of the CTS 3.9.8.2 Action is to ensure that radioactive material does not escape the containment should the SDC requirements continue to not be met

and boiling occurs in the core. Therefore, containment penetrations are closed to seal the containment. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering a reasonable time for repairs or replacement of required features, and the low probability of an accident occurring during the repair period. The Required Actions are consistent with the actions taken for containment closure in CTS 3.9.4 (i.e., ITS 3.9.3). Penetrations which can be closed by an OPERABLE Containment Isolation System do not need to be closed if SDC is inoperable, since the presence of radioactivity in the containment will cause the valves to close automatically, thus performing the isolation function. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.9 REFUELING OPERATIONS

3.9.5 Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level

3.9.8.2

LCO 3.9.5 Two SDC loops shall be OPERABLE, and one SDC loop shall be in operation.

----------NOTES----------

DOC L01

1. All SDC pumps may be removed from operation for ≤ 15 minutes when switching from one ~~train~~ loop to another provided: 1

   a. The core outlet temperature is maintained >10 degrees F below saturation temperature,

   b. No operations are permitted that would cause introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1, "Boron Concentration." and 2 2

   c. No draining operations to further reduce RCS water volume are permitted.

3.9.8.2 Footnote *

2. One required SDC loop may be inoperable for up to 2 hours for surveillance testing, provided that the other SDC loop is OPERABLE and in operation.

Applicability DOC M01

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.8.1 Action a. DOC M02 A. One SDC loop inoperable.	A.1 Initiate action to restore SDC loop to OPERABLE status. OR A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No SDC loop OPERABLE or in operation. (3.9.8.1 Action b. DOC M02)	B.1	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND		
(3.9.8.1 Action b.)	B.2	Initiate action to restore one SDC loop to OPERABLE status and to operation.	Immediately
	AND		
(DOC L02)	B.3	Close equipment hatch and secure with ~~[~~four~~]~~ bolts.	4 hours (3)
	AND		
(DOC L02)	B.4	Close one door in each air lock.	4 hours
	AND		
(3.9.8.1 Action b.)	B.5~~.1~~	Verify ~~Close~~ each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours (4)
	~~OR~~		
(DOC L02)	~~B.5.2~~	~~Verify each penetration~~ is capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Isolation System.	~~4 hours~~ (4) (1)

is either closed

; or

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.9.8.2.a	SR 3.9.5.1	Verify required SDC loops are OPERABLE and one SDC loop is in operation. and circulating reactor coolant at a flow rate of ≥ 3000 gpm	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 1 3
DOC M03	SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3
4.9.8.2.1	SR 3.9.5.3	Verify SDC loop locations susceptible to gas accumulation are sufficiently filled with water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~	3 3

3.9 REFUELING OPERATIONS

3.9.5 Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level

3.9.8.2

LCO 3.9.5 Two SDC loops shall be OPERABLE, and one SDC loop shall be in operation.

----------NOTES----------

DOC L01

1. All SDC pumps may be removed from operation for ≤ 15 minutes when switching from one ~~train~~ loop to another provided: 1

   a. The core outlet temperature is maintained >10 degrees F below saturation temperature,

   b. No operations are permitted that would cause introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1, "Boron Concentration." and 2

   c. No draining operations to further reduce RCS water volume are permitted. 2

3.9.8.2 Footnote **

2. One required SDC loop may be inoperable for up to 2 hours for surveillance testing, provided that the other SDC loop is OPERABLE and in operation.

Applicability

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.8.1 Action a. DOC M02 A. One SDC loop inoperable.	A.1 Initiate action to restore SDC loop to OPERABLE status. OR A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No SDC loop OPERABLE or in operation. (3.9.8.1 Action b. DOC M02)	B.1	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND		
(3.9.8.1 Action b.)	B.2	Initiate action to restore one SDC loop to OPERABLE status and to operation.	Immediately
	AND		
(DOC L02)	B.3	Close equipment hatch and secure with ~~[~~four~~]~~ bolts.	4 hours (3)
	AND		
(DOC L02)	B.4	Close one door in each air lock.	4 hours
	AND		
(3.9.8.1 Action b.)	B.5~~.1~~	~~Close~~ each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours (4)
	~~OR~~		
(DOC L02)	~~B.5.2~~	~~Verify each penetration~~ is capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Isolation System.	~~4 hours~~ (4) (1)

Verify

is either closed

; or

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
4.9.8.2.a, b	SR 3.9.5.1	Verify required SDC loops are OPERABLE and one SDC loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm. ----------NOTE---------- Reactor coolant flow rate may be reduced to ≥ 1850 gpm when the reactor is subcritical ≥ 125 hours, the maximum RCS temperature is ≤ 117°F, and component cooling water temperature to the SDC heat exchanger is ≤ 87°F. ----------	~~[12 hours~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
DOC M03	SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation.	~~[7 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~
4.9.8.2.c	SR 3.9.5.3	Verify SDC loop locations susceptible to gas accumulation are sufficiently filled with water.	~~[31 days~~ ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.5, SHUTDOWN COOLING (SDC) AND COOLANT CIRCULATION – LOW WATER LEVEL

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. Change made to be consistent with the ISTS format as specified in the Writer's Guide for the Improved Standard Technical Specifications, TSTF-GG-05-01 and other Specifications; for example, LCO 3.8.2.a reference to LCO 3.8.10, “Distribution Systems – Shutdown.”

3. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

4. ISTS 3.9.5 Required Actions B.5.1 and B.5.2 are connected by an "OR" logical connector, such that either one can be performed to meet the requirements of the ACTION. However, the two Required Actions as presented in the ISTS are applicable to all the containment penetrations; either Required Action B.5.1 or Required Action B.5.2 must be performed. Thus, the ISTS presentation will not allow one penetration to be isolated by use of a manual valve and another penetration to be capable of being closed by an OPERABLE Containment Isolation System. This is not the intent of the requirement. The requirement is based on ISTS LCO 3.9.3, which requires each penetration to be either: a) closed by a manual or automatic isolation valve, blind flange, or equivalent; or b) capable of being closed by an OPERABLE Containment Isolation System. For consistency with the actual LCO requirement, ISTS 3.9.5 Required Actions B.5.1 and B.5.2 have been combined into a single Required Action in ITS 3.9.5 Required Action B.5. This will allow each of the applicable containment penetrations to be either isolated or capable of being isolated, as required, to ensure containment closure can be achieved.

5. **Unit 2 only**: A Note is added to ITS SR 3.9.5.1 that allows a reduction of reactor coolant flow rate when specific conditions are met consistent with the St. Lucie Plant Unit 2 current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.9 REFUELING OPERATIONS

B 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS)~~, as required by GDC 34,~~ to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water (CCW) System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System, as well as adjustments in CCW temperature and flow.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two ~~trains~~ loops of the SDC System are required to be OPERABLE, and one train is required to be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to prevent this challenge.

SDC and Coolant Circulation - Low Water Level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both SDC loops must be OPERABLE. Additionally, one loop of the SDC System must be in operation in order to provide: at least

a. Removal of decay heat, and

b. Mixing of borated coolant to minimize the possibility of a criticality~~, and~~

BASES

LCO (continued)

~~c. Indication of reactor coolant temperature~~.

This LCO is modified by two Notes. Note 1 permits the SDC pumps to be removed from operation for ≤ 15 minutes when switching from one train to another. The circumstances for stopping both SDC pumps are to be limited to situations when the outage time is short ~~[~~and the core outlet temperature is maintained > 10 degrees F below saturation temperature~~]~~. The Note prohibits boron dilution by introduction of coolant into the RCS with boron concentration less than that required to meet the minimum boron concentration of LCO 3.9.1, or draining operations when SDC forced flow is stopped.

, "Boron Concentration."

Note 2 allows one SDC loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

An OPERABLE SDC loop consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path ~~and to determine the low end temperature~~. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

at least one of

Management of gas voids is important to SDC System OPERABILITY.

Additionally, since the SDC System is a manually operated system (i.e., it is not automatically actuated), each SDC loop is OPERABLE if it can be manually aligned (remote or local) to the SDC mode.

Both SDC pumps may be aligned to the Refueling Water Storage Tank to support filling or draining the refueling cavity or for performance of required testing.

APPLICABILITY

Two SDC loops are required to be OPERABLE, and one SDC loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System. MODE 6 requirements, with a water level ≥ 23 ft above the reactor vessel flange, are covered in LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level."

ACTIONS

A.1 and A.2

If one SDC loop is inoperable, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and to operation, or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is established at ≥ 23 ft above the

BASES

ACTIONS (continued)

reactor vessel flange, the Applicability will change to that of LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level," and only one SDC loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no SDC loop is in operation or no SDC loops are OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no SDC loop is in operation or no SDC loops are OPERABLE, action shall be initiated immediately and continued without interruption to restore one SDC loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE SDC loops and one operating SDC loop should be accomplished expeditiously.

B.3, B.4~~, B.5.1,~~ and B.5~~.2~~ 4

If no SDC loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with ~~[~~four~~]~~ bolts, 2

b. One door in each air lock must be closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Isolation System. 1

BASES

ACTIONS (continued)

With SDC loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions stated above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most SDC problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one SDC loop is operating and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, this Surveillance demonstrates that the other SDC loop is OPERABLE.

In addition, during operation of the SDC loop with the water level in the vicinity of the reactor vessel nozzles, the SDC loop flow rate determination must also consider the SDC pump suction requirements.

~~[The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the SDC System in the control room.~~ 2

~~Verification that the required loops are OPERABLE and in operation ensures that loops can be placed in operation as needed, to maintain decay heat and retain forced circulation. The Frequency of 12 hours is considered reasonable, since other administrative controls are available and have proven to be acceptable by operating experience.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.5.2

Verification that the required pump is OPERABLE ensures that an additional SDC pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

SR 3.9.5.3

SDC System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the SDC loops and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

Selection of SDC System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The SDC System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the SDC System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

SDC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the SDC System piping and the procedural controls governing system operation.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~

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REFERENCES 1. FSAR, Section ~~[~~ ~~]~~.

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15.2.4

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B 3.9 REFUELING OPERATIONS

B 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS)~~, as required by GDC 34,~~ to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water (CCW) System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System, as well as adjustments in CCW temperature and flow.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two ~~trains~~ loops of the SDC System are required to be OPERABLE, and one train is required to be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to prevent this challenge.

SDC and Coolant Circulation - Low Water Level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both SDC loops must be OPERABLE. Additionally, at least one loop of the SDC System must be in operation in order to provide:

a. Removal of decay heat, and

b. Mixing of borated coolant to minimize the possibility of a criticality~~, and~~

BASES

LCO (continued)

~~c. Indication of reactor coolant temperature~~.

This LCO is modified by two Notes. Note 1 permits the SDC pumps to be removed from operation for ≤ 15 minutes when switching from one train to another. The circumstances for stopping both SDC pumps are to be limited to situations when the outage time is short ~~[~~and the core outlet temperature is maintained > 10 degrees F below saturation temperature~~]~~. The Note prohibits boron dilution by introduction of coolant into the RCS with boron concentration less than that required to meet the minimum boron concentration of LCO 3.9.1, "Boron Concentration," or draining operations when SDC forced flow is stopped.

Note 2 allows one SDC loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

An OPERABLE SDC loop consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path ~~and to determine the low end temperature~~. The flow path starts in one of the RCS hot legs and is returned to at least one of the RCS cold legs. Management of gas voids is important to SDC System OPERABILITY.

Additionally, since the SDC System is a manually operated system (i.e., it is not automatically actuated), each SDC loop is OPERABLE if it can be manually aligned (remote or local) to the SDC mode.

Both SDC pumps may be aligned to the Refueling Water Storage Tank to support filling or draining the refueling cavity or for performance of required testing.

APPLICABILITY

Two SDC loops are required to be OPERABLE, and one SDC loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System. MODE 6 requirements, with a water level ≥ 23 ft above the reactor vessel flange, are covered in LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level."

ACTIONS

A.1 and A.2

If one SDC loop is inoperable, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and to operation, or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is established at ≥ 23 ft above the

BASES

ACTIONS (continued)

reactor vessel flange, the Applicability will change to that of LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level," and only one SDC loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no SDC loop is in operation or no SDC loops are OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no SDC loop is in operation or no SDC loops are OPERABLE, action shall be initiated immediately and continued without interruption to restore one SDC loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE SDC loops and one operating SDC loop should be accomplished expeditiously.

B.3, B.4~~, B.5.1,~~ and B.5~~.2~~ 4

If no SDC loop is in operation, the following actions must be taken:

a. The equipment hatch must be closed and secured with ~~[~~four~~]~~ bolts, 2

b. One door in each air lock must be closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment ~~Purge and Exhaust~~ Isolation System. 1

BASES

ACTIONS (continued)

With SDC loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions stated above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most SDC problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one SDC loop is operating and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, this Surveillance demonstrates that the other SDC loop is OPERABLE.

In addition, during operation of the SDC loop with the water level in the vicinity of the reactor vessel nozzles, the SDC loop flow rate determination must also consider the SDC pump suction requirements.

~~[The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the SDC System in the control room.~~ 2

~~Verification that the required loops are OPERABLE and in operation ensures that loops can be placed in operation as needed, to maintain decay heat and retain forced circulation. The Frequency of 12 hours is considered reasonable, since other administrative controls are available and have proven to be acceptable by operating experience.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~---------- REVIEWER'S NOTE ----------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~----------]~~ 2

The SR is modified by a Note that provides an allowance to reduce the required RCS flow rate when specific conditions are met. The time after reactor shutdown, RCS temperature, and CCW temperature to the shutdown cooling heat exchanger are initial conditions to ensure a reduced SDC flow rate will not result in a temperature transient exceeding the established refueling operational temperature limit. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.5.2

Verification that the required pump is OPERABLE ensures that an additional SDC pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. ~~[The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ 2

~~OR~~ 2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2
~~]~~

SR 3.9.5.3

SDC System piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the SDC loops and may also prevent water hammer, pump cavitation, and pumping of noncondensible gas into the reactor vessel.

Selection of SDC System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrumentation drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walk downs to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as stand-by versus operating conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The SDC System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If the accumulated gas is eliminated or brought within the acceptance criteria limits during performance of the Surveillance, the Surveillance is met and past system OPERABILITY is evaluated under the Corrective Action Program. If it is determined by subsequent evaluation that the SDC System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

SDC System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative sub-set of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, the plant configuration, or personnel safety. For these locations alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

~~[The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the SDC System piping and the procedural controls governing system operation.~~

2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequency may vary by location susceptible to gas accumulation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~[REVIEWER'S NOTE~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~]~~

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REFERENCES 1. FSAR, Section ~~[~~ ~~]~~.

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15.4.6

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JUSTIFICATION FOR DEVIATIONS
ITS 3.9.5, BASES, SHUTDOWN COOLING (SDC) AND COOLANT CIRCULATION – LOW WATER LEVEL

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Change made to be consistent with the ISTS format as specified in the Writer's Guide for the Improved Standard Technical Specifications, TSTF GG 05 01 and other ISTS Bases (e.g., ISTS Bases 3.9.1 refers to LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level").

4. Changes are made to reflect changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.5, SHUTDOWN COOLING (SDC) AND COOLANT CIRCULATION – LOW WATER LEVEL

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 6

3.9.6, Refueling Water Level

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

REFUELING OPERATIONS

WATER LEVEL – REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

LCO 3.9.6

~~3.9.10~~ ~~At least~~ ≥ 23 feet of water refueling level shall be maintained ~~over~~ above the top of ~~irradiated fuel assemblies seated within the~~ reactor ~~pressure~~ vessel flange. M01

Applicability

APPLICABILITY: ~~During CORE ALTERATIONS.~~ L01

During movement of irradiated fuel assemblies within containment.

ACTION:

ACTION A

Refueling water level not within limit.

~~With the requirements of the above specifications not satisfied~~, immediately suspend ~~CORE ALTERATIONS~~ ~~and~~ movement of irradiated fuel assemblies within containment~~, and immediately initiate action to restore refueling cavity water level to within limits~~. L01 A02

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verify refueling

is ≥ 23 feet above the top of reactor vessel flange M01

~~4.9.10~~ ~~The~~ water level ~~shall be determined to be at least its minimum required depth within 2 hours prior to the start of~~ ~~and~~ in accordance with the Surveillance Frequency Control Program ~~during CORE ALTERATIONS~~ ~~and during movement of irradiated fuel assemblies within containment~~. L02 L01 A03

A01

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL – REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

LCO 3.9.6

~~3.9.10~~ ~~At least~~ ≥ 23 feet of refueling level water shall be maintained ~~over~~ above the top of ~~the~~ reactor ~~pressure~~ vessel flange.

Applicability

APPLICABILITY: ~~During CORE ALTERATIONS.~~ During movement of irradiated fuel assemblies within containment. L01

ACTION:

ACTION A

~~With the requirements of the above specifications not satisfied~~, immediately suspend ~~CORE ALTERATIONS~~ ~~and~~ movement of irradiated fuel assemblies within containment~~, and immediately initiate action to restore refueling cavity water level to within limits~~. L01 A02

Refueling water level not within limit.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

~~4.9.10~~ ~~The~~ Verify refueling water level is ≥ 23 feet above the top of reactor vessel flange ~~shall be determined to be at least its minimum required depth within 2 hours prior to the start of~~ and in accordance with the Surveillance Frequency Control Program thereafter ~~during CORE ALTERATIONS~~ ~~and during movement of irradiated fuel assemblies within containment~~. L02 L01 A03

DISCUSSION OF CHANGES
ITS 3.9.6, REFUELING WATER LEVEL

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 3.9.10 requires the refueling cavity water level to be within the specified limit. CTS 3.9.10 Action requires immediate action to suspend CORE ALTERATIONS and movement of irradiated fuel assemblies in containment; and immediately initiate action to restore the refueling cavity water level to within the limit. ITS 3.9.6 ACTIONS do not include the action to initiate action to restore refueling cavity water level to within the limit. This changes the CTS by removing an unnecessary action. The purpose of the CTS 3.9.10 actions is to provide appropriate remedial actions when an LCO is not met consistent with the requirements of 10 CFR 50.36(c)(2)(i). The action to suspend CORE ALTERATIONS and movement of irradiated fuel assemblies within containment results in the LCO no longer being required (Refer to Discussion of Change L01 for removal of CORE ALTERATION Applicability and Action). CTS 3.0.2 (ITS LCO 3.0.2) state, in part, that when an LCO is met or is no longer applicable prior to expiration of the specified time interval(s), completion of the ACTIONS is not required, unless otherwise stated. As indicated in the ISTS Bases of ITS LCO 3.0.2, “Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.” Therefore, it is unnecessary to provide an action to specifically require the LCO to be met, as this is inherent in LCO 3.0.1 and LCO 3.0.2.

This change is designated as an administrative change and is acceptable because it does not result in technical changes to the CTS.

A03 CTS 4.9.10 requires the refueling cavity water level to be determined to be within limit in accordance with the Surveillance Frequency Control Program during movement of irradiated fuel assemblies within containment. ITS SR 3.9.6.1 requires verification that the refueling water level is within limit in accordance with the Surveillance Frequency Control Program. The CTS and ITS applicability are also during movement of irradiated fuel assemblies within containment. This changes the CTS by deleting the Applicability requirement from the Surveillance. ITS LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification). Additionally, SR 3.0.1 establishes the requirement that Surveillance Requirements must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. Therefore, maintaining the Applicability requirement in the Surveillance is not necessary.

This change is designated as an administrative change and is acceptable because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only**: CTS 3.9.10 requires a minimum of 23 feet of water be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel. ITS 3.9.6 requires 23 feet of water be maintained above the top of the reactor vessel flange. This changes the CTS by increasing the amount of water that must be in the refueling cavity during irradiated fuel movement.

Refueling water level is required to ensure the consequences of a design basis refuel accident remain within the bounds of the radiological dose calculations. Since the fuel handling accident could occur anywhere in the refueling cavity, the water level in the reactor vessel and refueling cavity must be at least 23 feet above the top of the reactor vessel flange. Therefore, the increased water level requirement is acceptable. This change is made consistent with the requirements of PSL Unit 2 CTS and the ISTS. This change is designated as more restrictive because it adds new requirements to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 2 – Relaxation of Applicability)* CTS 3.9.10 is applicable during CORE ALTERATIONS and movement of irradiated fuel assemblies within the containment. ITS 3.9.3 is applicable during movement of irradiated fuel assemblies within containment. References to CORE ALTERATIONS in CTS 3.9.4 are eliminated in the Applicability, Action, and Surveillances. This changes the CTS by eliminating requirements for maintaining refueling water level during CORE ALTERATIONS. | R1

The purpose of CTS 3.9.4 is to ensure the refueling water level satisfies the condition assumed in the Fuel Handling Accident (FHA) inside containment analysis. This change is acceptable because the requirements continue to ensure that the structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. CORE ALTERATIONS is defined in CTS 1.9, in part, as "the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity, within the reactor vessel with the head removed and fuel in the vessel." There is only one accident considered that

involves a CORE ALTERATION: An FHA. According to the Unit 1 UFSAR Section 15.4.3 and Unit 2 UFSAR Section 15.7.4.1.2, an FHA is initiated by the dropping of an irradiated fuel assembly either in the containment or in the fuel building. Suspension of CORE ALTERATIONS, except for suspension of movement of irradiated fuel, will not prevent or impair the mitigation of an FHA.

This change is acceptable because the likelihood of dropping a reactivity component other than an irradiated fuel assembly is minimized by following the regulatory guidelines of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” which includes establishing safe load paths, establishing and maintaining procedures for load-handling operations; training of crane operators; design, testing, inspection, and maintenance of cranes and lifting devices. The PSL crane design precludes the handling of heavy objects, such as shipping casks, over the reactor vessel. Administrative controls that control of movement of light loads or prevent movement of light loads over irradiated fuel assemblies are similar to those used for control of heavy loads, to the extent practicable, as advised in NUREG-0612. Consequently, the possibility of dropping a reactivity component other than an irradiated fuel assembly and damaging of fuel assemblies in the reactor vessel is remote. Therefore, imposing requirements during CORE ALTERATIONS in addition to during movement of irradiated fuel is unnecessary. This change is consistent with the ISTS and designated as less restrictive because the ITS LCO requirements are applicable in fewer operating conditions than in the CTS.

L02 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.9.10 requires the refueling cavity water level to be determined to be within limit "within 2 hours prior to the start of" and in accordance with the Surveillance Frequency Control Program (i.e., at least once per 24 hours). ITS SR 3.9.6.1 requires verification that the refueling water level is ≥ 23 feet above the top of the reactor vessel flange in accordance with the Surveillance Frequency Control Program (i.e., at least once per 24 hours). This changes the CTS by reducing the Frequency for verifying refueling water level from 2 hours before entering the Applicability of the LCO to 24 hours before entering the Applicability of the LCO.

The purpose of CTS 4.9.10 is to ensure that the refueling water level is greater than or equal to that assumed in the fuel handling accident analysis. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that the refueling water level is verified on a periodicity considering the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely. The Frequency of 24 hours is sufficient during the movement of fuel assemblies; therefore, it is sufficient before fuel assemblies are moved. CTS 4.0.4 (ITS SR 3.0.4) requires the SR to be met within its Frequency requirements prior to entering the MODE or other specified conditions in the Applicability except as provided by CTS 4.0.3 (ITS SR 3.0.3 – missed SR). The reactor cavity minimum water level must be met prior to commencing movement of irradiated fuel assemblies or fuel assembly movement must be suspended immediately (thereby exiting the Applicability of the Specification). Therefore, changing the Frequency from 2 hours before moving irradiated fuel assemblies to within 24 hours before moving irradiated fuel assemblies has no effect on plant safety. This change is

designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

3.9 REFUELING OPERATIONS

3.9.6 Refueling Water Level

3.9.10

LCO 3.9.6 Refueling water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

Applicability

APPLICABILITY: During movement of ~~[recently]~~ irradiated fuel assemblies within containment. (2)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.10 Action A. Refueling water level not within limit.	A.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies within containment.	Immediately (2)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.9.10 SR 3.9.6.1	Verify refueling water level is ≥ 23 ft above the top of reactor vessel flange.	~~[24 hours~~ (2) ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ (2)

3.9 REFUELING OPERATIONS

3.9.6 Refueling Water Level

3.9.10

LCO 3.9.6 Refueling water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

Applicability

APPLICABILITY: During movement of ~~[recently]~~ irradiated fuel assemblies within containment. 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.10 Action A. Refueling water level not within limit.	A.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies within containment.	Immediately 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.9.10 SR 3.9.6.1	Verify refueling water level is ≥ 23 ft above the top of reactor vessel flange.	~~[24 hours~~ 2 ~~OR~~ In accordance with the Surveillance Frequency Control Program ~~]~~ 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.6, REFUELING WATER LEVEL

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling this maintains sufficient water level in the containment, ~~the refueling canal,~~ the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (FHA) (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~< 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.~~ within the radiation dose criteria set forth in 10 CFR 50.67, as provided by the guidance of Reference 1

1 R1 1

APPLICABLE SAFETY ANALYSES

~~During movement of irradiated fuel assemblies, the water level in the refueling canal and refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).~~

INSERT 1

1

~~The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of [X] hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).~~

1

Refueling water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A minimum refueling water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated ~~fuel handling accident~~ FHA inside containment are within acceptable limits as provided by the guidance of Reference 3.

1

APPLICABILITY

LCO 3.9.6 is applicable when moving fuel assemblies in the presence of irradiated fuel assemblies. The LCO minimizes the possibility of ~~a fuel handling accident~~ an FHA in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated ~~fuel handling accident~~ FHA. Requirements for ~~fuel handling accidents~~ FHAs in the spent fuel pool are covered by LCO 3.7.~~10~~ 13, "Fuel Storage Pool Water Level." Spent

1 1 3

1

1

INSERT 1

The FHA analysis is described in Reference 2. During movement of irradiated fuel assemblies, the water level in the reactor cavity is an initial condition design parameter in the analysis of the FHA in containment postulated in Appendix B or Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 feet is maintained above the damaged fuel assembly for both the containment and fuel handling building (FHB) release locations. Fuel handling accidents consist of dropping a single irradiated fuel assembly either in the FHB or inside the containment and assumes all the fuel rods in the assembly are damaged. The offsite dose analysis does not credit filtration during an FHA inside containment with the maintenance hatch open or in the spent fuel storage pool. The control room dose analysis credits control room emergency ventilation filtration during an FHA inside or outside the containment. R1

In the event of an FHA, a minimum water level of 23 feet and a minimum decay time of 72 hours prior to fuel handling ensures an effective iodine decontamination factor of 200 per the guidance provided in NRC Regulatory Issue Summary 2006-04 (Ref. 3). The FHA assumes 100% of the noble gas is released from the damaged fuel assembly and escapes from the reactor cavity. All of the non-iodine particulates released from the damaged fuel assembly are assumed to be retained by the water. Iodine released from the damaged fuel assembly is assumed to be composed of 99.85% elemental and 0.15% organic. Activity released from the FHA is assumed to leak to the environment over a two-hour period and no credit is taken for dilution in the containment.

BASES

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the postulated ~~fuel~~ FHA ~~handling accident~~ analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from ~~a~~ an FHA ~~fuel handling accident~~ inside containment (Ref. 2). 1

~~[The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972~~ 1.183, July 2000. 1
2. FSAR, Section ~~[ ]~~ 15.4.3 U 2
3. ~~NUREG-0800, Section 15.7.4.~~ 1
4. ~~10 CFR 100.10~~. NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternate Source Terms," March 7, 2006. 1

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling this maintains sufficient water level in the containment, ~~the refueling canal,~~ the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (FHA) (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~< 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.~~ within the radiation dose criteria set forth in 10 CFR 50.67, as provided by the guidance of Reference 1

R1

1

1

APPLICABLE SAFETY ANALYSES

~~During movement of irradiated fuel assemblies, the water level in the refueling canal and refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).~~

INSERT 1

1

~~The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of [X] hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).~~

1

Refueling water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A minimum refueling water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated ~~fuel handling accident~~ FHA inside containment are within acceptable limits as provided by the guidance of Reference 3.

APPLICABILITY

LCO 3.9.6 is applicable when moving fuel assemblies in the presence of irradiated fuel assemblies. The LCO minimizes the possibility of ~~a fuel handling accident~~ an FHA in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated ~~fuel handling accident~~ FHA. Requirements for ~~fuel handling accidents~~ FHAs in the spent fuel pool are covered by LCO 3.7.~~10~~13, "Fuel Storage Pool Spent Water Level."

1

1

3

1

1

INSERT 1

The FHA analysis is described in Reference 2. During movement of irradiated fuel assemblies, the water level in the reactor cavity is an initial condition design parameter in the analysis of the FHA in containment postulated in Appendix B or Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 feet is maintained above the damaged fuel assembly for both the containment and fuel handling building (FHB) release locations. Fuel handling accidents consist of dropping a single irradiated fuel assembly either in the FHB or inside the containment and assumes all the fuel rods in the assembly are damaged. The offsite dose analysis does not credit filtration during an FHA inside containment with the maintenance hatch open or in the spent fuel storage pool. The control room dose analysis credits control room emergency ventilation filtration during an FHA inside or outside the containment. R1

In the event of an FHA, a minimum water level of 23 feet and a minimum decay time of 72 hours prior to fuel handling ensures an effective iodine decontamination factor of 200 per the guidance provided in NRC Regulatory Issue Summary 2006-04 (Ref. 3). The FHA assumes 100% of the noble gas is released from the damaged fuel assembly and escapes from the reactor cavity. All of the non-iodine particulates released from the damaged fuel assembly are assumed to be retained by the water. Iodine released from the damaged fuel assembly is assumed to be composed of 99.85% elemental and 0.15% organic. Activity released from the FHA is assumed to leak to the environment over a two-hour period and no credit is taken for dilution in the containment.

BASES

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the postulated ~~fuel~~ FHA ~~handling accident~~ analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a an FHA ~~fuel handling accident~~ inside containment (Ref. 2). 1

~~[The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.~~ 2

~~OR~~

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

~~REVIEWER'S NOTE~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~ 2

~~]~~

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972~~ 1.183, July 2000. 1
2. FSAR, Section ~~[ ]~~ 15.4.3 U 2
3. ~~NUREG-0800, Section 15.7.4.~~ 1
4. ~~10 CFR 100.10~~. NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternate Source Terms," March 7, 2006. 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.6, BASES, REFUELING WATER LEVEL

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. Changes have been made to be consistent with title and number changes made to other Specifications.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.9.6, REFUELING WATER LEVEL

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

R1

Relocated/Deleted Current Technical Specifications (CTS)

CTS 3/4.9.3, DECAY TIME

R1

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

R1

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for a minimum of 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

LA01

R1

~~**REFUELING OPERATIONS**~~

~~**3/4.9.3 DECAY TIME**~~

~~**LIMITING CONDITION FOR OPERATION**~~

~~3.9.3 The reactor shall be subcritical for at least 72 hours.~~

~~**APPLICABILITY:** During movement of irradiated fuel in the reactor pressure vessel.~~

~~**ACTION:**~~

~~With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.~~

~~**SURVEILLANCE REQUIREMENTS**~~

~~4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.~~

DISCUSSION OF CHANGES
CTS 3/4.9.3, DECAY TIME

ADMINISTRATIVE CHANGES R1

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program) CTS 3/4.9.3 provides requirements associated with Decay Time. Specifically, CTS 4.9.3 requires a determination of verifying that the reactor has been subcritical for at least 72 hours by a verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for less than 72 hours, the CTS 3.9.3 Action requires suspension of all operations involving movement of irradiated fuel in the reactor pressure vessel (RPV). ITS does not include a requirement for decay time. This changes the CTS by moving the explicit decay time requirements from the Technical Specifications to the Technical Requirements Manual (TRM).

The purpose of CTS 3/4.9.3 is to ensure that sufficient time has elapsed to allow radioactive decay of the short-lived fission products in the irradiated fuel consistent with the assumptions used in the fuel handling accident analysis. The removal of this administrative control detail from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. It is improbable to move irradiated fuel within 72 hours from entering MODE 3 (i.e., keff < 0.99) because of the physical time required to perform plant shutdown, cooldown, depressurize the Reactor Coolant System (RCS), and the additional operations required prior to moving recently irradiated fuel in the reactor vessel (e.g., containment entry, removal of vessel head, removal of vessel internals, etc.). Therefore, movement of irradiated fuel prior to the 72 hour decay is precluded. Thus, it is unnecessary to retain the decay time requirement in Technical Specifications. ITS retains Specifications to mitigate a fuel handling accident associated with the movement of recently irradiated fuel, which encompasses the unlikely movement of fuel prior to a decay period of 72 hours. Specifically, Specifications associated with the following systems will ensure these systems are OPERABLE during movement of recently irradiated fuel:

- engineered safety feature ventilation systems, the Unit 1 fuel building ventilation system, and associated instrumentation actuation functions;

R1

- control room emergency ventilation system and associated instrumentation actuation functions;
- reactor vessel and spent fuel pool minimum water level;
- containment penetration requirements and associated containment isolation instrumentation; and
- electrical systems needed to support systems listed herein.

The administrative requirement to determine that the reactor has been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the RPV will be relocated to the TRM, along with the action requirement to immediately suspend irradiated fuel movement in the unlikely event that irradiated fuel movement did occur < 72 hours from subcriticality. Any change to the decay time input assumption in the fuel handling accident analysis will be evaluated pursuant to the criteria of 10 CFR 50.59 c(2). This change is acceptable because the removed information will be adequately controlled in the TRM. Changes to the TRM are controlled by the provisions of 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a requirement is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.9.3, DECAY TIME

R1

There are no specific No Significant Hazards Considerations for this Specification.

ENCLOSURE 2

VOLUME 15

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 4.0
DESIGN FEATURES

Revision 1

ATTACHMENT 1

ITS Chapter 4.0, Design Features

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

~~SECTION 5.0~~

~~DESIGN FEATURES~~

5.0 DESIGN FEATURES

5.1 SITE LOCATION

Plant

4.1
Insert 1

The St. Lucie nuclear units are located on Hutchinson Island in St. Lucie County, about halfway between the cities of Fort Pierce and Stuart on the east coast of Florida. The radius of the exclusion area is 0.97 miles from the center of the St. Lucie Plant. The low population zone ~~(LPZ)~~ includes that area within one mile of the center of the St. Lucie Plant.

a radius of

5.2 CONTAINMENT

LA01

CONFIGURATION

5.2.1 The containment structure is comprised of a steel containment vessel, having the shape of a right circular cylinder with a hemispherical dome and ellipsoidal bottom, surrounded by a reinforced concrete shield building. The radius of the shield building is at least 4 feet greater than the radius of circular cylinder portion of the containment vessel at any point.

5.2.1.1 CONTAINMENT VESSEL

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 232 feet.

c. Net free volume = 2.5 x 10^6 cubic feet.

d. Nominal thickness of vessel walls = 2 inches.

e. Nominal thickness of vessel dome = 1 inch.

f. Nominal thickness of vessel bottom = 2 inches.

~~Figure 5.1-1 Deleted~~

~~FIGURE 5.1-2~~
~~(Deleted)~~

DESIGN FEATURES

LA01

2.1.2 SHIELD BUILDING

a. Minimum annular space = 4 feet

b. Annulus nominal volume = 543,000 cubic feet

c. Nominal outside height (measured from top of foundation base to the top of the dome) = 230.5 feet

d. Nominal inside diameter = 148 feet

e. Cylinder wall minimum thickness = 3 feet

f. Dome minimum thickness = 2.5 feet

g. Dome inside radius = 112 feet

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment vessel is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 264°F.

PENETRATIONS

5.2.3 Penetrations through the containment structure are designed and shall be maintained in accordance with the original design provisions contained in Sections 3.8.2.1.10 and 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

4.2

5.3 REACTOR CORE

FUEL ASSEMBLIES

4.2.1

5.3.1 The reactor core shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or M5 clad fuel rods ~~and/or poison rods~~, with fuel rods having an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

LA02

~~5.3.1.1 Except for special test as authorized by the NRC, all fuel assemblies under control element assemblies shall be sleeved with a sleeve design previously approved by the NRC.~~

DESIGN FEATURES

CONTROL ELEMENT ASSEMBLIES

4.2.2 5.3.2 The reactor core shall contain 73 ~~full length and no part length~~ control element assemblies. The control element assemblies shall be designed ~~and maintained in accordance with the original design provisions contained in Section 4.2.3.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.~~ with control material of silver indium cadmium or boron carbide as approved by the NRC.

A02 LA02 M01

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

LA01

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

4.3 ### 5.6 FUEL STORAGE

are designed and

Add ITS 4.3.1.1.a

M02

CRITICALITY

4.3.1.1 5.6.1.a The spent fuel ~~pool and spent fuel~~ storage racks shall be maintained with:

4.3.1.1.b 1. k_{eff} less than 1.0 when fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

4.3.1.1.d
4.3.1.1.e
4.3.1.1.f 2. A nominal 10.12 inches center to center distance between fuel assemblies in Region 1 of the spent fuel pool storage racks, a nominal 10.30 inches center to center distance between fuel assemblies in the Region 1 cask pit storage rack, and a nominal 8.86 inches center to center distance between fuel assemblies in Region 2 of the spent fuel pool storage racks.

4.3.1.1.c
Insert 2 3. A k_{eff} less than or equal to 0.95 when flooded with water containing 500 ppm boron, including an allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

DESIGN FEATURES

CRITICALITY (Continued)

A04

4. For storage of enriched fuel assemblies, requirements of Criteria in 5.6.1.a.1 and 5.6.1.a.3 shall be met by positioning fuel in the spent fuel storage racks consistent with the requirements of Specification 5.6.1.c.

~~5. Vessel Flux Reduction Assemblies (VFRAs), as defined in Section 9.1 of the Updated Final Safety Analysis Report, may be placed in any allowable fuel storage location.~~

LA03

~~6. Fissile material, not contained in a fuel assembly lattice, shall be stored in accordance with the requirements of Criteria in 5.6.1.a.1 and 5.6.1.a.3.~~

A03

~~7. The Metamic neutron absorber inserts shall have a ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$.~~

LA03

~~b. The Region 1 cask pit storage rack shall contain neutron absorbing material (Boral) between stored fuel assemblies when installed in the spent fuel pool.~~

c. Loading of spent fuel storage racks shall be controlled as described below. Criteria in 5.6.1.c.2, 5.6.1.c.3, 5.6.1.c.5 and 5.6.1.c.6 do not apply to the Region 1 cask pit storage rack.

1. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel storage rack shall be less than or equal to 4.6 weight percent.

See ITS 3.7.15

2. Fuel placed in Region 1 of the spent fuel pool storage racks shall comply with the storage patterns and alignment restrictions of Figure 5.6-1 and the minimum burnup requirements of Table 5.6-1.

3. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage patterns or allowed special arrangements of Figure 5.6-2 and the minimum burnup requirements of Table 5.6-1. The allowed special arrangement for fresh fuel may be repeated, provided the applicable interface requirements specified by the safety analysis are met.

4. Any fuel satisfying criteria 5.6.1.c.1, including fresh fuel, may be placed in the Region 1 cask pit storage rack.

5. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required.

6. Any 2x2 array of Region 2 storage cells that interface with Region 1 shall comply with the rules of Figure 5.6-3. The allowed special arrangement in Region 2 as shown in Figure 5.6-2 shall not be placed adjacent to Region 1.

4.3.1.2.a
4.3.1.2.b
4.3.1.2.c

d. The new fuel storage racks ~~are designed for dry storage of unirradiated fuel assemblies~~ having a maximum planar average U-235 enrichment less than or equal to 4.6 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 ~~under the most reactive condition~~.

LA02

if fully flooded with unborated water or if moderated by aqueous foam including allowances for uncertainties as described in UFSAR Section 9.1

M02

Add ITS 4.3.1.2.d

DESIGN FEATURES

DRAINAGE

4.3.2 5.6.2 The fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

CAPACITY

4.3.3 5.6.3 The spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1706 fuel assemblies, and the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 143 fuel assemblies. The total Unit 1 spent fuel pool and cask pit storage capacity is limited to no more than 1849 fuel assemblies.

LA01

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as seismic Class I in Section 3.2.1 of the FSAR shall be designed and maintained to the original design provisions contained in Section 3.7 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirement.

~~5.8 DELETED~~

~~5.9 DELETED~~

~~DELETED~~

~~SECTION 5.0~~

~~DESIGN FEATURES~~

5.0 DESIGN FEATURES

5.1 SITE LOCATION

~~EXCLUSION AREA~~

Plant

4.1
Insert 1

The St. Lucie nuclear units are located on Hutchinson Island in St. Lucie County, about halfway between the cities of Fort Pierce and Stuart on the east coast of Florida. The radius of the exclusion area is 0.97 miles from the center of the St. Lucie Plant. The low population zone ~~(LPZ)~~ includes that area within one mile of the center of the St. Lucie Plant.

a radius of

5.2 CONTAINMENT

LA01

CONFIGURATION

5.2.1 The reactor containment building is a steel building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 232 feet.

c. Net free volume = 2.506×10^6 cubic feet.

d. Nominal thickness of vessel walls = 2 inches.

e. Nominal thickness of vessel dome = 1 inch.

f. Nominal thickness of vessel bottom = 2 inches.

5.2.1.2 SHIELD BUILDING

a. Minimum annular space = 4 feet.

b. Annulus nominal volume = 543,000 cubic feet.

c. Nominal outside height (measured from top of foundation mat to the top of the dome) = 228.5 feet.

d. Nominal inside diameter = 148 feet.

e. Cylinder wall minimum thickness = 3 feet.

f. Dome minimum thickness = 2.5 feet.

g. Dome inside radius = 112 feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The steel reactor containment building is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 264°F.

~~Figure 5.1-1 Deleted~~

DESIGN FEATURES

4.2 **5.3 REACTOR CORE**

FUEL ASSEMBLIES

4.2.1 5.3.1 The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO™ or M5® clad fuel rods ~~and/or poison rods~~, with fuel rods having an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

LA02

CONTROL ELEMENT ASSEMBLIES

4.2.2 5.3.2 The reactor core shall contain 87 ~~full-length~~ control element assemblies ~~and no part-length control element assemblies~~.

A02

The control material shall be silver indium cadmium or boron carbide as approved by the NRC.

M01

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

LA01

DESIGN FEATURES

~~5.5 DELETED~~

4.3

5.6 FUEL STORAGE

CRITICALITY

Add ITS 4.3.1.1.a

M02

4.3.1.1 5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

4.3.1.1.b

1. A k_{eff} equivalent to less than 1.0 when flooded with unborated water, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

4.3.1.1.c Insert 2

2. A k_{eff} equivalent to less than or equal to 0.95 when flooded with water containing 500 ppm boron, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

4.3.1.1.d 4.3.1.1.e 4.3.1.1.f

3. A nominal 8.965 inch center-to-center distance between fuel assemblies placed in the spent fuel pool storage racks and a nominal 8.80 inch center-to-center distance between fuel assemblies placed in the cask pit storage rack.

Region 1 and 2

Region 2

~~4. For storage of enriched fuel assemblies, requirements of Specification 5.6.1.a.1 and 5.6.1.a.2 shall be met by positioning fuel in the spent fuel pool storage racks consistent with the requirements of Specification 5.6.1.c.~~

~~5. Fissile material, not contained in a fuel assembly lattice, shall be stored in accordance with the requirements of Specifications 5.6.1.a.1 and 5.6.1.a.2.~~

A03

~~6. The Metamic neutron absorber inserts shall have a ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$.~~

LA03

~~b. The cask pit storage rack shall contain neutron absorbing material (Boral) between stored fuel assemblies when installed in the spent fuel pool.~~

c. Loading of spent fuel pool storage racks shall be controlled as described below.

See ITS 3.7.15

1. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel pool storage rack shall be less than or equal to 4.6 weight percent.

2. Fuel placed in Region 1 of the spent fuel pool storage racks shall comply with the storage pattern definitions of Figure 5.6-1 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)

3. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions or allowed special arrangement definitions of Figure 5.6-2 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)

DESIGN FEATURES (continued)

CRITICALITY (continued)

See ITS 3.7.15

5.6.1 c. (continued)

4. The 2x2 array of fuel assemblies that span the interface between Region 1 and Region 2 of the spent fuel pool storage racks shall comply with the storage pattern definitions of Figure 5.6-3 and the minimum burnup requirements as defined in Table 5.6-1. The allowed special arrangements in Region 2 as shown in Figure 5.6-2 shall not be placed adjacent to Region 1. (See Specification 5.6.1.c.7 for exceptions)

5. Fuel placed in the cask pit storage rack shall comply with the storage pattern definitions of Figure 5.6-4 and the minimum burnup requirements as defined in Table 5.6-1. (See Specification 5.6.1.c.7 for exceptions)

6. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required.

7. Fresh or spent fuel in any allowed configuration may be replaced with non-fuel hardware, and fresh fuel in any allowed configuration may be replaced with a fuel rod storage basket containing fuel rod(s). Also, storage of Metamic inserts or control rods, without any fissile material, is acceptable in locations designated as completely water-filled cells.

4.3.1.2.a
4.3.1.2.b
4.3.1.2.c

d. The new fuel storage racks ~~are designed for dry storage of unirradiated fuel assemblies~~ having a maximum planar average U-235 enrichment less than or equal to 4.6 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 ~~under the most reactive condition~~.

LA02

if fully flooded with unborated water or if moderated by aqueous foam including allowances for uncertainties as described in UFSAR Section 9.1.

DRAINAGE

4.3.2 5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

Add ITS 4.3.1.2.d

M02

CAPACITY

4.3.3 5.6.3 The spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1491 fuel assemblies, and the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 225 fuel assemblies. The total Unit 2 spent fuel pool and cask pit storage capacity is limited to no more than 1716 fuel assemblies.

~~**5.7 DELETED**~~

~~Pages 5-4C through 5-4F (Amendment 101) and page 5-4G (Amendment 135) have been deleted from the Technical Specifications. The next page is 5-4h.~~

A01

~~DELETED~~

A01

~~DELETED~~

A01

~~DELETED~~

A01

~~DELETED~~

DISCUSSION OF CHANGES
ITS 4.0, DESIGN FEATURES

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 5.3.2 includes the descriptor “full length” and “no part length” in reference to control element assemblies (CEAs).

The purpose of the CTS requirement is to describe the number of CEAs and the control material used in the CEA design. Unit 1 and Unit 2 no longer use partial length CEA(s). Since partial length CEA(s) are no longer a design feature for Unit 1 and Unit 2, it is unnecessary to describe CEA types that PSL does not have. Therefore, the “full length” and “no partial length” descriptors used for CEA(s) are deleted. This change is designated as administrative and is acceptable because it does not result in a technical change to the CTS.

A03 Unit 1 CTS 5.6.1.a.6 requires compliance with CTS 5.6.1.a.1 and a.3. Unit 2 CTS 5.6.1.a.4 requires compliance with CTS 5.6.1.a.1 and a.2 when positioning fuel in the spent fuel pool storage racks consistent with the requirements of CTS 5.6.1.c. Unit 2 CTS 5.6.1.a.5 requires compliance with CTS 5.6.1.a.1 and a.2. This changes the CTS by not including a specific requirement to comply with another CTS requirement.

The purpose of CTS 5.6.1 requirements is to ensure the spent fuel storage racks are designed and maintained to comply with 10 CFR 50.68, “Criticality accident requirements.” Since the spent fuel storage racks are designed, as stated in CTS 5.6.1.a.1, a.2, and a.3 and Section 9.1 of the Updated Safety Analysis Report (UFSAR), to comply with the requirements of 10 CFR 50.68, it is unnecessary to restate that other CTS 5.6.1 Specifications shall be met when positioning fuel assemblies in the spent fuel storage racks and when storing fissile material not contained in a fuel assembly lattice. It is not an ITS convention to include this type of references to comply with other Technical Specification requirements. ITS 4.3.1.1 continues to require spent fuel storage racks be designed to comply with 10 CFR 50.68 regardless of whether fissile material is contained in a fuel assembly lattice or not and the requirements provided in CTS 5.6.1.c (ITS 3.7.15) continue to assure control of fuel storage in the spent fuel storage pool. This change is designated as administrative because unnecessarily duplicative requirements to comply with other Technical Specification requirements are eliminated with no technical change to the CTS.

A04 **Unit 1 only**: CTS 5.6.1.a.4 states that the spent fuel storage rack criteria of CTS 5.6.1.a.1 and 5.6.1.a.3 shall be met for storage of enriched fuel assemblies by positioning fuel in the spent fuel storage racks consistent with the requirements of Specification 5.6.1.c. ITS 4.3.1.1 does not include this statement of fact. The

purpose of CTS 5.6.1.a.4 is to ensure enriched fuel assemblies are stored consistent with the pattern and location requirements specified in CTS 5.6.1.c to ensure the criticality requirements of CTS 5.6.1.a.1 and 5.6.1.a.3 are met. Compliance with the requirements of CTS 5.6.1.c are required for irradiated fuel assemblies, including enriched fuel assemblies. The location and pattern requirements of CTS 5.6.1.c ensure the requirements of CTS 5.6.1.a.1 and 5.6.1.a.3 are met. Therefore, it is unnecessary to include an explicit requirement for enriched fuel assemblies to be stored in a specific manner to ensure criticality limits are met. This change is designated as administrative because an unnecessarily duplicative requirement to comply with other Specification requirements is eliminated with no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 ITS 4.2.2 requires the control material of the CEAs to consist of silver indium cadmium or boron carbide as approved by the NRC. The CTS does not include explicit detail on the type of CEA control material that must be used. This changes the CTS by adding specific CEA control material requirements.

The purpose of the CEA design feature requirement is to provide information considered as a design feature description as described in 10 CFR 50.36(c)(4). The specific CEA control material is considered a CEA material of construction, which, if altered or modified, would have a significant effect on safety. This change is designated more restrictive because design feature information is added to the CTS consistent with the ISTS.

M02 CTS 5.6.1.a requires the spent fuel storage racks to be maintained within specific criticality requirements; however, CTS does not specify U-235 enrichment for irradiated fuel assemblies. ITS 4.3.1.1.a requires fuel assemblies in the spent fuel storage racks (i.e., spent fuel pool and cask pit storage racks) to be limited to a maximum U-235 enrichment of 4.6 weight percent. This changes the CTS by adding a specific fuel assembly U-235 enrichment limit.

CTS 5.6.1.d requires new fuel storage racks, in part, to maintain a k_{eff} less than or equal to 0.98 under the most reactive conditions. ITS 4.3.1.2.b and c require new fuel storage racks to be designed and maintained with a $k_{eff} \leq 0.98$ if fully flooded with unborated water and if moderated by aqueous foam, respectively, including allowance for uncertainties described in Section 9.1 of the UFSAR. ITS 4.3.1.2.d also requires that new fuel storage racks are designed and maintained to have a nominal 21 inch, for Unit 1, and 23 inch, for Unit 2, center to center distance between fuel assemblies in order to prevent criticality of the fuel assemblies. This changes the CTS by adding specific reactivity conditions and fuel assembly spacing requirements for the design and maintenance of the new fuel storage racks.

The purpose of ITS 4.3.1.1 is to prevent criticality in the spent fuel storage pool and cask pit. U-235 enrichment is an important input in spent fuel pool and cask pit criticality analyses.

The purpose of ITS 4.3.1.2 is to prevent criticality in the new fuel storage vault. Unit 1 UFSAR Section 9.1.1.1 states that the new fuel storage racks are designed, in part, to provide sufficient spacing between the fuel assemblies to maintain a subcritical array during flooding with nonborated water. Unit 2 UFSAR Section 9.1.1.1 states that the new fuel storage racks are designed, in part, to provide sufficient spacing between the fuel assemblies to maintain a subcritical (Keff ≤ 0.98) array assuming the most reactive condition. The establishment of minimum spacing design requirements for the new fuel storage racks assists in meeting this design basis and provides assurance that no incident could occur that would result in a hazard to public health and safety. The specific new fuel keff criteria is added consistent with 10 CFR 50.68(b)(2) and (b)(3).

This change is acceptable because it provides appropriate limits for the new and spent fuel storage racks and is designated more restrictive because general design requirements in CTS have been replaced with specific requirements in ITS consistent with the ISTS and 10 CFR 50.68.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 and Unit 2 CTS 5.2 and 5.4 provide description details of the containment systems and the reactor coolant system (RCS), respectively. Unit 1 CTS 5.5 and 5.7 provides description details of the emergency core cooling systems (ECCS) and seismic classifications, respectively. This changes the CTS by moving the description details of these systems to the UFSAR.

The removal of these details, which are related to system description, from the Technical Specifications, is acceptable because this type of information is not considered a design feature requirement as described in 10 CFR 50.36(c)(4); and therefore, is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains requirements on containment systems OPERABILITY in ITS Section 3.6, “Containment Systems,” RCS OPERABILITY in ITS Section 3.4, “Reactor Coolant Systems (RCS),” and ECCS OPERABILITY in ITS Section 3.5, “Emergency Core Cooling Systems (ECCS).” Seismic classification requirements continue to be required as promulgated by regulation; specifically, Appendix B, Section III, Design Control, and Appendix S, Section IV, Application To Engineering Design, of 10 CFR 50. Also, this change is acceptable because the information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made per the provisions of 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA02 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS and Unit 2 CTS 5.3.1 state that each fuel assembly, in part, consists of "and/or poison rods." In addition, Unit 1 CTS 5.3.1.1 states that, except for special test as authorized by the NRC, fuel assemblies under CEAs must be sleeved with an NRC approved sleeve design. Unit 1 CTS 5.3.2 contains reference to the original CEA design provisions in the FSAR and allows for normal degradation allowance pursuant to applicable Surveillance Requirements. Unit 1 and Unit 2 CTS 5.6.1.d explicitly state that the new fuel storage racks are designed for dry storage of unirradiated fuel. ITS 4.2.1, 4.2.2, and 4.3.1.2 do not include this level of detail and is revised consistent with the ISTS. This changes the CTS by moving this level of detail to the UFSAR.

The removal of these details, which are related to system design, from the CTS is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retain fuel assembly and CEA material description (refer to DOC M01), which, if altered or modified, would have a significant effect on safety. Additionally, ITS retains requirements on CEA OPERABILITY in Section 3.1. ITS also retains sufficient criticality information associated with the new fuel storage racks, which, if altered or modified, would have a significant effect on safety. These requirements provide adequate assurance the fuel assemblies, CEAs, and new fuel storage racks are capable of performing their safety function. Additionally, the removed information is more appropriately contained in the UFSAR. The inclusion of the details of component design and material in the UFSAR is consistent with the content and purpose of the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures that changes to the material contained in the UFSAR are properly evaluated. Therefore, the removal of this information from the CTS and placement in the UFSAR is acceptable. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the CTS.

LA03 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 5.6.1.a.5 allows Vessel Flux Reduction Assemblies (VFRAs), as defined in Section 9.1 of the Updated Final Safety Analysis Report, to be placed in any allowable fuel storage location. Unit 1 CTS 5.6.1.a.7 and Unit 2 CTS 5.6.1.a.6 require the Metamic neutron absorber inserts to have a ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$ and Unit 1 and Unit 2 CTS 5.6.1.b requires the cask pit storage rack to contain neutron absorbing material (Boral) between stored fuel assemblies when installed in the spent fuel pool. ITS 4.3.1.1 does not include this level of detail and is revised consistent with the ISTS. This changes the CTS by moving this level of detail to the UFSAR.

The removal of these details, which are related to specific rack design, from the CTS is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains sufficient criticality information associated with the spent fuel and cask pit storage racks, which, if altered or modified, would have a significant effect on safety. The removed information is more appropriately contained in the UFSAR. The inclusion of the details of component design and material in the UFSAR is consistent with the content and purpose of

the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures that changes to the material contained in the UFSAR are properly evaluated. Therefore, the removal of this information from the CTS and placement in the UFSAR is acceptable. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the CTS.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

4.0 DESIGN FEATURES

5.1 ## 4.1 Site Location

INSERT 1

~~[Text description of site location.]~~

2

5.3 ## 4.2 Reactor Core

5.3.1 ### 4.2.1 Fuel Assemblies

The reactor shall contain ~~[~~217~~]~~ fuel assemblies. Each assembly shall consist of a matrix of ~~[~~Zircalloy or ~~ZIRLO]~~ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

M5 clad

2

Element

5.3.2 ### 4.2.2 ~~[~~Control ~~Rod]~~ Assemblies

73

DOC M01

The reactor core shall contain ~~[91]~~ control element assemblies (CEAs). The control material shall be ~~[~~silver indium cadmium~~,~~ boron carbide~~, or hafnium metal]~~ as approved by the NRC.

or

2

5.6 ## 4.3 Fuel Storage

5.6.1.a ### 4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

DOC M02

4.6

planar average

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ weight percent,

1

2

< 1.0

b. k_{eff} ~~≤ 0.95~~ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[~~Section 9.1 of the ~~FSAR]~~,

3

Updated Final Safety Analysis Report (UFSAR)

INSERT 2

10.12

Region 1 spent fuel pool

~~[c~~. A nominal ~~[9]~~ inch center to center distance between fuel assemblies placed in ~~[~~the ~~high density fuel~~ storage racks~~]~~, ~~]~~

d

8.86

Region 2 spent fuel pool

~~[d~~. A nominal ~~[10.4]~~ inch center to center distance between fuel assemblies placed in ~~[~~the ~~low density fuel~~ storage racks~~]~~, ~~]~~

e

2

and

3

INSERT 3

2

Amendment XXX

1

2

INSERT 1

5.1 The St. Lucie Plant nuclear units are located on Hutchinson Island in St. Lucie County, about halfway between the cities of Fort Pierce and Stuart on the east coast of Florida. The radius of the exclusion area is 0.97 miles from the center of the St. Lucie Plant. The low population zone is the area within a radius of one mile from the center of the St. Lucie Plant.

INSERT 2 3

5.6.1.a.3 c. $k_{eff} \leq 0.95$ if flooded with borated water at a soluble boron concentration of 500 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

INSERT 3 2

5.6.1.a.2 f. A nominal 10.30 inch center to center distance between fuel assemblies placed in the Region 1 cask pit storage rack.

CTS

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

~~[e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure [3.7.18-1] may be allowed unrestricted storage in [either] fuel storage rack(s), and]~~

~~[f. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure [3.7.18-1] will be stored in compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]~~

4

5.6.1.d

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

planar average

1

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ weight percent,

4.6

5.6.1.d
DOC M02

b. $k_{eff} \leq 0.98$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[~~Section 9.1 of the FSAR~~]~~,

U

5.6.1.d
DOC M02

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in ~~[~~Section 9.1 of the FSAR~~]~~, and

U

21

DOC M02

d. A nominal ~~[10]~~ inch center to center distance between fuel assemblies placed in the storage racks.

2

5.6.2 4.3.2 Drainage

56

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation ~~[23~~ ft~~]~~.

5.6.3 4.3.3 Capacity

and cask pit are

total

1

The spent fuel storage pool~~s~~ is designed and shall be maintained with a storage capacity limited to no more than ~~[1542]~~ fuel assemblies.

1849

with the spent fuel pool storage racks limited to no more than 1706 fuel assemblies and the cask pit storage rack limited to no more than 143 fuel assemblies

1

CTS

4.0 DESIGN FEATURES

5.1 ## 4.1 Site Location

INSERT 1

~~[Text description of site location.]~~

2

5.3 ## 4.2 Reactor Core

5.3.1 ### 4.2.1 Fuel Assemblies

The reactor shall contain ~~[~~217~~]~~ fuel assemblies. Each assembly shall consist of a matrix of ~~[~~Zircalloy~~,~~ ~~or~~ ZIRLO~~]~~ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

, or M5 clad

2

Element

5.3.2 ### 4.2.2 ~~[~~Control ~~Rod]~~ Assemblies

87

DOC M01

The reactor core shall contain ~~[91]~~ control element assemblies (CEAs). The control material shall be ~~[~~silver indium cadmium~~,~~ boron carbide~~, or hafnium metal]~~ as approved by the NRC.

or

2

5.6 ## 4.3 Fuel Storage

4.3.1 Criticality

5.6.1.a 4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

DOC M02

4.6

planar average

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ weight percent,

1

2

< 1.0

b. k_{eff} ~~≤ 0.95~~ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[~~Section 9.1 of the ~~FSAR]~~,

3

Updated Final Safety Analysis Report (UFSAR)

INSERT 2

8.965

Region 1 spent fuel pool

~~[c~~. A nominal ~~[9]~~ inch center to center distance between fuel assemblies placed in ~~[~~the ~~high density fuel~~ storage racks~~]~~, ~~]~~

d

8.965

Region 2 spent fuel pool

~~[d~~. A nominal ~~[10.4]~~ inch center to center distance between fuel assemblies placed in ~~[~~the ~~low density fuel~~ storage racks~~]~~, ~~]~~

e

and

2

3

INSERT 3

2

Amendment XXX

1

CTS

2

INSERT 1

5.1 The St. Lucie Plant nuclear units are located on Hutchinson Island in St. Lucie County, about halfway between the cities of Fort Pierce and Stuart on the east coast of Florida. The radius of the exclusion area is 0.97 miles from the center of the St. Lucie Plant. The low population zone is the area within a radius of one mile from the center of the St. Lucie Plant.

INSERT 2 3

5.6.1.a.2 c. $k_{eff} \leq 0.95$ if flooded with borated water at a soluble boron concentration of 500 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

INSERT 3 2

5.6.1.a.3 f. A nominal 8.8 inch center to center distance between fuel assemblies placed in the Region 2 cask pit storage rack.

CTS

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

~~[e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure [3.7.18-1] may be allowed unrestricted storage in [either] fuel storage rack(s), and]~~

~~[f. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure [3.7.18-1] will be stored in compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]~~

4

5.6.1.d

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

1

planar average

a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ weight percent,

4.6

5.6.1.d
DOC M02

b. $k_{eff} \leq 0.98$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[~~Section 9.1 of the FSAR~~]~~,

U

5.6.1.d
DOC M02

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in ~~[~~Section 9.1 of the FSAR~~]~~, and

U

23

DOC M02

d. A nominal ~~[10]~~ inch center to center distance between fuel assemblies placed in the storage racks.

2

5.6.2

4.3.2 Drainage

56

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation ~~[23~~ ft~~]~~.

5.6.3

4.3.3 Capacity

1

and cask pit are

total

The spent fuel storage pool ~~is~~ designed and shall be maintained with a storage capacity limited to no more than ~~[1542]~~ fuel assemblies.

1716

with the spent fuel pool storage racks limited to no more than 1491 fuel assemblies and the cask pit storage rack limited to no more than 225 fuel assemblies

1

JUSTIFICATION FOR DEVIATIONS
ITS 4.0, DESIGN FEATURES

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The St. Lucie Plant (PSL) spent fuel storage criticality analyses for Units 1 and 2 indicate that to maintain keff ≤ 0.95 under normal conditions, boron concentration must be at least 500 ppm, including an allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report (UFSAR). Additionally, if fully flooded with unborated water, spent fuel storage criticality analyses indicate keff will remain below 1.0, which includes an allowance for biases and uncertainties as described in UFSAR, Section 9.1. Therefore, these changes are made to reflect the PSL fuel storage criticality safety analyses.

4. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The bracketed information is not included in the PSL Unit 1 and Unit 2 ITS. The bracketed information is not related to the spent fuel storage rack design, but rather is related to storage allowances of specific fuel assemblies. PSL is adopting ISTS 3.7.18, “Spent Fuel Storage” (ITS 3.7.15), which defines the plant specific requirements for location of fuel assemblies in the spent fuel storage pool. In addition, specific requirements associated with fuel assembly storage are provided in UFSAR Section 9.1.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 4.0, DESIGN FEATURES

There are no specific No Significant Hazards Considerations for this Chapter.

ENCLOSURE 2

VOLUME 16

ST. LUCIE PLANT
UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 5.0
ADMINISTRATIVE CONTROLS

Revision 1

LIST OF ATTACHMENTS

1. ITS Section 5.1, Responsibility
2. ITS Section 5.2, Organization
3. ITS Section 5.3, Unit Staff Qualifications
4. ITS Section 5.4, Procedures
5. ITS Section 5.5, Programs and Manuals
6. ITS Section 5.6, Reporting Requirements
7. ITS Section 5.7, High Radiation Area
8. Relocated/Deleted Current Technical Specifications (CTS)

ATTACHMENT 1

ITS Section 5.1, Responsibility

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ITS

A01

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

ITS 5.1

6.0 ADMINISTRATIVE CONTROLS

5.2 **6.1 RESPONSIBILITY**

M01

5.1.1 6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

the

5.1.2 6.1.2 The Shift Supervisor shall be responsible for the control room command function. During any absence of the Shift Supervisor from the control room while ~~either~~ unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while ~~both units are~~ in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

L01

the unit is

6.2 ORGANIZATION

See ITS 5.2

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for unit operation and corporate management. This onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3).
The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR or the Quality Assurance Topical Report.

b. A specified corporate officer shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The plant manager shall be responsible for overall safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

ITS

A01

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

ITS 5.1

6.0 ADMINISTRATIVE CONTROLS

5.2

6.1 RESPONSIBILITY

M01

5.1.1

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the

5.1.2

6.1.2 The Shift Supervisor shall be responsible for the control room command function. During any absence of the Shift Supervisor from the control room while ~~either~~ unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while ~~both units are~~ in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

L01

the unit is

6.2 ORGANIZATION

See ITS 5.2

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for unit operation and corporate management. This onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3).
The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR or the Quality Assurance Topical Report.

b. A specified corporate officer shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

DISCUSSION OF CHANGES
ITS 5.1, RESPONSIBILITY

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 CTS 6.1.1 provides requirements related to responsibility of the plant manager but does not contain any information concerning the plant manager's (or designee's) role in plant activities that affect nuclear safety. ITS 5.1.1 requires the plant manager or his designee to approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety. This changes the CTS by requiring the plant manager or his designee to approve certain activities to systems and equipment, prior to implementation, that affect nuclear safety.

The purpose of CTS 6.1.1 is to provide specific responsibilities for the plant manager. The proposed change adds a requirement for the plant manager (or his designee) to approve, prior to implementation, proposed tests, experiments, and modifications to systems or equipment that affect nuclear safety. The addition of this responsibility will not alter any of the existing responsibilities. The requirement is being added to ensure the plant manager is cognizant of those activities to plant systems or equipment that can potentially affect nuclear safety. This change is designated as more restrictive because it adds an additional responsibility for the plant manager in the administrative controls chapter of the ITS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L01 *(Category 1 – Relaxation of LCO Requirements)* CTS 6.1.2 requires an individual with an active Senior Reactor Operator (SRO) license to provide control room command function during any absence of the Shift Supervisor from

the control room while either unit is in MODE 1, 2, 3, or 4. An individual with an active Reactor Operator (RO) license may also assume the control room command function during any absence of the Shift Supervisor from the control room while both units are in MODE 5 or 6. ITS 5.1.2 provides similar requirements except an individual with an active RO license may assume the control room command function during the absence of the Shift Supervisor while the unit associated with the control room is in MODE 5 or 6 irrespective of the operational status of the other unit. This changes the CTS by relaxing the control room command function requirement from requiring an SRO to assume these responsibilities while either unit is in MODE 1, 2, 3, or 4 to requiring the SRO to assume the control room command function only when the associated unit is in MODE 1, 2, 3, or 4 and allowing the RO to assume these responsibilities when the unit associated with the control room is in MODE 5 or 6.

The purpose of CTS 6.1.2 is to ensure compliance with the 10 CFR 50.54(m)(iii) requirement that, when a unit is in an operational mode other than cold shutdown or refueling, a person holding a senior operator license for the unit is present at the controls at all times and that when the unit is fueled, including cold shutdown and refueling operational modes, either a senior licensed operator or a licensed operator is present at the controls at all times. Each PSL unit is controlled from a separate control room. The CTS requirement is written for a two unit site with a common control room to ensure an SRO maintains the control room command function when either unit is in MODES 1, 2, 3, and 4. However, with separate and independent control rooms, the assignment of the control room command function is based on the operational status of the associated unit and is irrespective of the operational status of other unit.

This change is acceptable because the ITS continues to assure that a senior licensed operator (i.e., Shift Supervisor or SRO) maintains control room command function in the control room when the associated unit is in MODE 1, 2, 3, or 4 and either a senior licensed operator or a licensed operator (i.e., RO) maintains control room command function in the control room when the associated unit is shutdown and still fueled (i.e., in MODES 5 and 6). Therefore, ITS 5.1.2 will continue to satisfy the requirements of 10 CFR 50.54(m)(iii). Additionally, this change does not alter the minimum emergency response organizational requirements of the PSL emergency plan and therefore, does not impact PSL compliance with 10 CFR 50.45, Emergency plan, or 10 CFR 50, Appendix E, Emergency Planning and Preparedness for Production and Utilization Facilities. This change is designated as less restrictive because less stringent requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

6.1 5.1 Responsibility

~~REVIEWER'S NOTES~~

~~1. Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special titles because of unique organizational structures.~~

~~2. The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title as apply with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.~~

2

6.1.1 5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

DOC M01 The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

shift supervisor

3

6.1.2 DOC L01 5.1.2 The ~~[~~Shift Supervisor ~~(SS)]~~ shall be responsible for the control room command function. During any absence of the ~~[SS]~~ from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

6.1 ## 5.1 Responsibility

~~REVIEWER'S NOTES~~

~~1. Titles for members of the unit staff shall be specified by use of an overall statement referencing an ANSI Standard acceptable to the NRC staff from which the titles were obtained, or an alternative title may be designated for this position. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special titles because of unique organizational structures.~~

2

~~2. The ANSI Standard shall be the same ANSI Standard referenced in Section 5.3, Unit Staff Qualifications. If alternative titles are used, all requirements of these Technical Specifications apply to the position with the alternative title as apply with the specified title. Unit staff titles shall be specified in the Final Safety Analysis Report or Quality Assurance Plan. Unit staff titles shall be maintained and revised using those procedures approved for modifying/revising the Final Safety Analysis Report or Quality Assurance Plan.~~

6.1.1 5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

DOC M01 The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

shift supervisor

3

6.1.2 DOC L01 5.1.2 The ~~[~~Shift Supervisor ~~(SS)]~~ shall be responsible for the control room command function. During any absence of the ~~[SS]~~ from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

JUSTIFICATION FOR DEVIATIONS
ITS 5.1, RESPONSIBILITY

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal. Titles used in ITS Section 5.1 align with the titles used in ANSI Standard ANSI/ANS 3.1-1978, “Standard for Selection and Training of Personnel for Nuclear Power Plants,” and alternate titles are not used. ANSI/ANS 3.1-1978 is the same ANS Standard referenced in ITS Section 5.3 (CTS 6.3).

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.1, RESPONSIBILITY

There are no specific No Significant Hazards Considerations for this Section.

ATTACHMENT 2

ITS Section 5.2, Organization

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor shall be responsible for the control room command function. During any absence of the Shift Supervisor from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while both units are in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.2 ## 6.2 ORGANIZATION

See ITS 5.1

ONSITE AND OFFSITE ORGANIZATION

5.2.1 6.2.1 ~~An o~~Onsite and ~~an~~ offsite organizations shall be established for unit operation and corporate management. The~~is~~ onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

, respectively

5.2.1.a a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in ~~the form of~~ organizational charts. ~~These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3).~~ The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR or the Quality Assurance Topical Report.

A06

LA01

A02

, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation

5.2.1.c b. A specified corporate officer shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

5.2.1.b c. The plant manager shall be responsible for overall safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

; however

5.2.1.d d. ~~Although the i~~Individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite; they shall have sufficient organizational freedom to be independent from operating pressures.

5.2.1.d e. ~~Although h~~Health physics individuals may report to any appropriate manager onsite, ~~for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.~~

; however, these individuals shall have sufficient freedom to ensure their operational independence.

LA02

6.0 ADMINISTRATIVE CONTROLS

5.2 **6.2 ORGANIZATION** (continued)

UNIT STAFF

5.2.2 6.2.2 The unit staff organization shall ~~meet the requirements of 10 CFR 50.54(m) and~~ include the following: A02

5.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4. A ~~minimum~~ of three non-licensed operators is required when both units are in MODE~~S~~ 5 or 6. A01

for both units

total

5.2.2.b b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.2.a and 6.2.2.e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

radiation protection

A03

5.2.2.c c. A ~~health physics~~ technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5.2.2.d d. The operations manager or assistant operations manager shall hold an SRO license.

5.2.2.e e. An individual ~~(Shift Technical Advisor (STA))~~ shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. ~~The STA position shall be manned in MODES 1, 2, 3, or 4. by use of either a dedicated STA, a Shift Supervisor who meets the qualifications for the STA as required by Technical Specification 6.3.1, or an individual assigned to the unit with a Senior Reactor Operator's license who meets the qualifications for the STA as required by Technical Specification 6.3.1. If the STA position is filled by an STA qualified Shift Supervisor or dedicated STA, then the~~ individual may ~~fill the same position on Unit 2~~.

A04

A04

A05

A04

~~DELETED~~

~~DELETED~~

~~DELETED~~

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor shall be responsible for the control room command function. During any absence of the Shift Supervisor from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while both units are in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.2 ## 6.2 ORGANIZATION

See ITS 5.1

ONSITE AND OFFSITE ORGANIZATION

5.2.1 6.2.1 ~~An o~~Onsite and ~~an~~ offsite organizations shall be established for unit operation and corporate management. The~~is~~ onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

, respectively

5.2.1.a a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in ~~the form of~~ organizational charts. ~~These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3).~~ The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR or the Quality Assurance Topical Report.

, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation

A06

LA01

A02

5.2.1.c b. A specified corporate officer shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

5.2.1.b c. The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

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5.2.1.d d. ~~Although the i~~Individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite; they shall have sufficient organizational freedom to be independent from operating pressures.

5.2.1.d e. ~~Although h~~Health physics individuals may report to any appropriate manager onsite, ~~for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.~~

; however, these individuals shall have sufficient freedom to ensure their operational independence.

LA02

6.0 ADMINISTRATIVE CONTROLS

5.2 **6.2 ORGANIZATION** (Continued)

UNIT STAFF

A02

5.2.2 6.2.2 The unit staff organization shall ~~meet the requirements of 10 CFR 50.54(m) and~~ include the following:

5.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4. A ~~minimum~~ of three non-licensed operators is required when both units are in MODE~~S~~ 5 or 6.

for both units

total

A01

5.2.2.b b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.2.a and 6.2.2.e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

A03

radiation protection

5.2.2.c c. A ~~health physics~~ technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5.2.2.d d. The operations manager or assistant operations manager shall hold an SRO license.

5.2.2.e e. An individual ~~(Shift Technical Advisor (STA))~~ shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. ~~The STA position shall be manned in MODES 1, 2, 3, or 4 by use of either a dedicated STA, a Shift Supervisor who meets the qualifications for the STA as required by Technical Specification 6.3.1, or an individual assigned to the unit with a Senior Reactor Operator's license who meets the qualifications for the STA as required by Technical Specification 6.3.1. If the STA position is filled by an STA qualified Shift Supervisor or dedicated STA, then the individual may fill the same position on Unit 1.~~

A04

A04

A05

A04

~~DELETED~~

~~DELETED~~

~~DELETED~~

DISCUSSION OF CHANGES
ITS 5.2, ORGANIZATION

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 6.2.1 requires, in part, to update the organizational charts documented in the Quality Assurance Topical Report (QATR) in accordance with 10 CFR 50.54(a)(3). CTS 6.2.2 requires, in part, for the unit staff organization to meet the requirements of 10 CFR 50.54(m). The ITS does not include these statements. This changes the CTS by removing the duplicate requirement to comply with the requirements of 10 CFR 50.54. Each PSL facility operating license requires the licensee to be subject to the conditions specified in 10 CFR Section 50.54. Therefore, it is unnecessary to state organizational charts in the QATR be updated in accordance with 10 CFR 50.54(a)(3) and the unit staff meet the requirements of 10 CFR 50.54(m) because it is duplicative. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 6.2.2.c requires, in part, that a health physics technician be on site when fuel is in the reactor. ITS 5.2.2.c requires a radiation protection technician to be on site when fuel is in the reactor. This changes the CTS by modifying the generic title of health physics technician with the more appropriate generic title of radiation protection technician. The purpose of CTS 6.2.2.c is to ensure there is an individual responsible for radiation protection on site when fuel is in the reactor. This change is consistent with the generic position in the ISTS and the current plant-specific title for these technicians. The generic title also aligns with the position title provided in ANSI/ANS 3.1-1978, “American National Standard for Selection and Training of Nuclear Power Plant Personnel.” This change is designated as administrative because it does not result in technical changes to the CTS.

A04 CTS 6.2.2.e requires an individual (Shift Technical Advisor (STA)) to provide advisory technical support to the unit operations shift crew, specifically be manned in MODES 1, 2, 3, or 4, and if the STA position is filled by an STA qualified Shift Supervisor or dedicated STA, then the individual may fill the same position on both units. ITS 5.2.2.e requires an individual to provide advisory technical support to the unit operations shift crew but is revised to eliminate reference to the STA position, when the position must be manned, and the details of who may fill STA role for both units. This changes the CTS by eliminating the inference that the STA role is a separate shift crew position instead of a function.

The purpose of the CTS requirement is to ensure engineering expertise is available on shift to provide advisory technical support. NRC Generic Letter

86-04 “Policy Statement on Engineering Expertise on Shift,” promulgated the policy statement regarding engineering expertise on shift and established the requirements for eliminating the separate STA position. The NRC policy statement offers the licensees two options for meeting the NUREG-0737, “Clarification of TMI Action Plan Requirements,” Enclosure 3, Item I.A.1.1 requirement regarding engineering expertise on shift and meeting licensed operator staffing requirements pursuant to 10 CFR 50.54(m)(2). Option 1 provides for elimination of the separate STA position by allowing licensees to combine one of the required SRO positions with the STA position into a dual-role (SRO/STA) function. Option 2 allows a licensee to continue use of an NRC-approved STA program while meeting licensed operator staffing requirements. Per the policy statement, the Commission encourages licensees to move toward the dual-role (SRO/STA) function, with the eventual goal of the shift supervisor serving in the dual role. Additionally, Item I.A.1.1 of NUREG-0737, Enclosure 3, allows the individual providing advisory technical support function to serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

ITS 5.2.2.e eliminates the title of "Shift Technical Advisor (STA)," because the on shift advisory technical support function may be fulfilled by one or more of the other on-shift individuals. This change is necessary so that it does not imply that the STA and the shift supervisor must be different individuals. As a result of eliminating the STA “position,” it is unnecessary to state when the STA “position” must be manned. In addition, Item I.A.1.1 of NUREG-0737, Enclosure 3, allows the individual providing advisory technical support to serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units and there is no regulatory requirement that conflicts with this statement. Therefore, it is unnecessary to restate this as an allowance in the Technical Specifications and does not prevent PSL from utilizing this allowance. This change is designated as administrative because an individual with engineering expertise (i.e., dedicated individual or dual role individual) will continue to be required on shift and this same individual may provide support to both units as allowed by NUREG-0737. Therefore, the change does not result in technical changes to the CTS.

A05 CTS 6.2.2.e, in part, includes a description of who specifically may provide advisory technical support to the unit operations shift crew: either a dedicated STA, a Shift Supervisor who meets the qualifications for the STA as required by Technical Specification 6.3.1, or an individual assigned to the unit with a Senior Reactor Operator's license who meets the qualifications for the STA as required by Technical Specification 6.3.1. ITS 5.2.2.e requires an individual to provide advisory technical support to the unit operations shift crew but is revised to eliminate the details of who is qualified to assume the duties. This changes the CTS by eliminating details regarding the individual assigned to provide technical advisory support to the unit operations shift crew.

The purpose of the CTS requirement is to ensure engineering expertise is available on shift to provide advisory technical support. NRC Generic Letter 86-04, promulgated the policy statement regarding engineering expertise on shift and established the requirements for eliminating the separate STA position. The NRC policy statement allowed for the elimination of the separate STA position by

allowing licensees to combine one of the required SRO positions with the STA position into a dual-role (SRO/STA) function. The policy statement indicates that either a dedicated STA, a Shift Supervisor who meets the qualifications or an individual assigned to the unit with a Senior Reactor Operator's license who meets the qualifications may serve the dual-role function.

It is an ITS convention to not include cross-references to comply with other Specification requirements. Compliance with other Specifications is understood. Compliance with CTS 6.3.1 (ITS 5.3.1) is required and referencing this Specification is unnecessary. Refer to ITS 5.3, “Unit Staff Qualifications,” for changes to CTS 6.3.1 requirements. This change is designated as administrative because an individual with engineering expertise and appropriate qualifications will continue to be required on shift. Therefore, the change does not result in technical changes to the CTS.

A06 CTS 6.2.1.a, regarding documentation and updating of the relationships between operating organization positions, requires that the lines of authority, responsibility and communication be established and defined from the highest management levels through intermediate levels to, and including, all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3). ITS 5.2.1.a states "Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation." This changes the CTS by further clarifying the methods of documenting the organizational relationships in the QA Topical Report. R1

The purpose of the CTS requirement is to ensure the organizational relationships are described and maintained in the QA Topical Report. This change clarifies methods of documenting the organizational relationships in the QA Topical Report and is acceptable because the ITS continues to require the organizational relationships documented and updated in the QA Topical Report. This change is designated as administrative because it includes clarification as to methods of documenting organizational relationships in the QA Topical Report.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 6.2.1.a requires, in part, to document the organizational charts in the QATR. ITS 5.2.1.a does not include this requirement. The requirement that organizational charts be documented in the QATR is moved to the Updated Final Safety Analysis Report (UFSAR). This changes the CTS by moving a requirement to the UFSAR. The removal of this requirement from the Technical Specifications is acceptable because this type of information is administrative and not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to document and update, in organizational charts, the relationship of the management levels through intermediate levels to and including operating organization positions, as appropriate. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because requirements for meeting Technical Specifications are being relocated from the Technical Specifications to the UFSAR.

LA02 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements*) CTS 6.2.1.e states that, although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures. ITS 5.2.1.d provides a similar requirement but does not provide procedural detail related to organizational freedom for health physics individuals (e.g., radiation manager shall have direct access to the individual having responsibility for overall unit management and health physics individuals' authority to cease any work activity when worker safety is jeopardized). ITS 5.2.1.d states, in part, "The individuals who… carry out health physics… functions… shall have sufficient organizational freedom to ensure their independence from operating pressures." The procedural detail of what activities constitute operational independence for health physics individuals, including the radiation protection manager's direct access to overall unit management individual, is moved to the UFSAR. This changes the CTS by moving procedural detail related to health physics individuals' responsibilities to the UFSAR and replacing the information with a general statement. The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement for health physics individuals to have sufficient freedom to ensure their operational independence. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

6.2 5.2 Organization

6.2.1 5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

6.2.1.a a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the ~~[FSAR/QA Plan]~~, (2)

Updated Final Safety Analysis Report or the Quality Assurance Topical Report

6.2.1.c b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant,

6.2.1.b c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety, and (4)

functions

6.2.1.d
6.2.1.e d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 5.2.2 Unit Staff

The unit staff organization shall include the following:

6.2.2.a a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

~~REVIEWER'S NOTE~~

~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~ (3)

A total of three non-licensed operators are required for both units when both units are in MODE 5 or 6.

5.2 Organization

6.2.2 5.2.2 Unit Staff (continued)

6.2.2.b b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

6.2.2.c c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

6.2.2.d d. The operations manager or assistant operations manager shall hold an SRO license.

6.2.2.e e. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

6.2 5.2 Organization

6.2.1 5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

6.2.1.a

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the ~~[FSAR/QA Plan]~~, (2)

Updated Final Safety Analysis Report or the Quality Assurance Topical Report

6.2.1.c

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant,

6.2.1.b

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety, and (4)

functions

6.2.1.d
6.2.1.e

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 5.2.2 Unit Staff

The unit staff organization shall include the following:

6.2.2.a

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

~~REVIEWER'S NOTE~~

~~Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.~~ (3)

A total of three non-licensed operators are required for both units when both units are in MODE 5 or 6.

5.2 Organization

6.2.2 5.2.2 Unit Staff (continued)

6.2.2.b b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

6.2.2.c c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

6.2.2.d d. The operations manager or assistant operations manager shall hold an SRO license.

6.2.2.e e. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

JUSTIFICATION FOR DEVIATIONS
ITS 5.2, ORGANIZATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal. Since the PSL is a two unit site, ISTS 5.2.2.a includes an additional statement requiring a total of three non-licensed operators for both units when both units are in MODE 5 or 6 (i.e., shutdown) consistent with the CTS requirement. This requirement does not extend to when both units are defueled because the ISTS nor 10 CFR 50.54(m) require licensed operators assigned to the reactor or control room when the reactor is defueled. Therefore, it unnecessary to require non-licensed operators when both units are defueled.

4. Grammatical correction made to the ITS to clarify ISTS 5.5.2.d includes individuals required to carry out health physics “functions.”

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.2, ORGANIZATION

There are no specific No Significant Hazards Considerations for this Section.

ATTACHMENT 3

ITS Section 5.3, Unit Staff Qualifications

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

6.0 ADMINISTRATIVE CONTROLS

5.3 **6.3 UNIT STAFF QUALIFICATIONS**

5.3.1 6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI / ANS-3.1-1978 for comparable positions, except for~~:~~ the radiation protection manager.

~~(1)~~ ~~t~~The radiation protection manager ~~who~~ shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975~~,~~. Revision 1,

~~(2) the Shift Technical Advisor who shall have specific training in plant design and plant operating characteristics, including transients and accidents, and any of the following educational requirements:~~

L01

- ~~Bachelor's degree in engineering from an accredited institution; or~~
- ~~Professional Engineer's (PE) license obtained by successful completion of the PE examination; or~~
- ~~Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences, or~~
- ~~Bachelor's degree in physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.~~

LA01

(3) the Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant.

c. Training: Complete the Multi-Discipline Supervisor training program.

5.3.2 6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of ~~6.3.1~~ Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

~~6.4 DELETED~~

~~6.5 DELETED~~

~~6.0 ADMINISTRATIVE CONTROLS~~

~~DELETED~~

~~6.0 ADMINISTRATIVE CONTROLS~~

~~DELETED~~

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~~ADMINISTRATIVE CONTROLS~~

~~6.6 DELETED~~

~~6.7 DELETED~~

6.0 ADMINISTRATIVE CONTROLS

5.3 6.3 UNIT STAFF QUALIFICATIONS

5.3.1 6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI / ANS-3.1-1978 for comparable positions, except for~~:~~ the radiation protection manager.

~~(1)~~ ~~t~~The radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975~~,~~. Revision 1,

~~(2) the Shift Technical Advisor who shall have specific training in plant design and plant operating characteristics, including transients and accidents, and any of the following educational requirements:~~

L01

- ~~Bachelor's degree in engineering from an accredited institution; or~~
- ~~Professional Engineer's (PE) license obtained by successful completion of the PE examination; or~~
- ~~Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences, or~~
- ~~Bachelor's degree in physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.~~

(3) the Multi-Discipline Supervisors who shall meet or exceed the following requirements: LA01

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant.

c. Training: Complete the Multi-Discipline Supervisor training program.

5.3.2 6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of ~~6.3.1~~, perform the functions described in 10 CFR 50.54(m).

Specification 5.3.1

~~6.0 ADMINISTRATIVE CONTROLS~~

~~6.4 DELETED~~

~~6.5 DELETED~~

~~6.0 ADMINISTRATIVE CONTROLS~~

~~DELETED~~

~~6.0 ADMINISTRATIVE CONTROLS~~

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DISCUSSION OF CHANGES
ITS 5.3, UNIT STAFF QUALIFICATIONS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program, or Surveillance Frequency Control Program)* CTS 6.3.1(3) provides minimum education, experience, and training requirements for multi-discipline supervisors. ITS 5.3.1 does not include these requirements. The minimum education, experience, and training requirements for multi-discipline supervisors are moved to the Updated Final Safety Analysis Report (UFSAR). This changes the CTS by moving an administrative requirement to the UFSAR. The removal of this requirement from the Technical Specifications is acceptable because this type of information is administrative and not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains minimum qualification requirements for members of the facility staff for comparable positions as defined in ANSI/ANS-3.1-1978 or Regulatory Guide 1.8, as appropriate. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because administrative requirements in Technical Specifications are being relocated from the Technical Specifications to the UFSAR.

LESS RESTRICTIVE CHANGES

L01 (*Category 1 – Relaxation of LCO Requirements*) CTS 6.3.1(2) provides specific training requirements and minimum educational requirements for the Shift Technical Advisor. ITS 5.3.1 does not include these requirements. This changes

the CTS by deleting explicit administrative training and qualification requirements for the shift technical advisor function.

The purpose of CTS 6.3.1(2) is to ensure the individual that provides advisory technical support to the unit operations shift crew (i.e., shift technical advisor) meets minimum qualifications and training pursuant to NUREG-0737, "Clarification of TMI Action Plan Requirements," Enclosure 3, Item I.A.1.1 This change is acceptable because the Florida Power & Light Company Quality Assurance Topical Report requires training for positions identified in 10CFR50.120, which includes the shift technical advisor position, be accomplished according to programs accredited by the National Nuclear Accrediting Board of the National Academy for Nuclear Training that implements a systematic approach to training. The systematic approach to training process develops and maintains training programs to ensure individuals are trained and meet minimum educational requirements necessary to perform their required tasks. Therefore, an individual who provides the shift technical advisor function will continue to be adequately trained and qualified to provide engineering expertise on shift. This change is designated as less restrictive because less stringent administrative LCO requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

5.0 ADMINISTRATIVE CONTROLS

6.3 ## 5.3 Unit Staff Qualifications

~~REVIEWER'S NOTE~~

~~Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~ (2)

6.3.1 5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff].~~ ~~[The staff not covered by Regulatory Guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff]~~. (3)

6.3.2 5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

ANSI/ANS-3.1-1978 for comparable positions, except for the radiation protection manager. The radiation protection manager shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 1, September 1975.

CTS

5.0 ADMINISTRATIVE CONTROLS

6.3 ## 5.3 Unit Staff Qualifications

~~REVIEWER'S NOTE~~

~~Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.~~ (2)

6.3.1 **5.3.1** Each member of the unit staff shall meet or exceed the minimum qualifications of ~~[Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff].~~ ~~[The staff not covered by Regulatory Guide 1.8 shall meet or exceed the minimum qualifications of Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff]~~. (3)

6.3.2 **5.3.2** For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of Specification 5.3.1, perform the functions described in 10 CFR 50.54(m).

ANSI/ANS-3.1-1978 for comparable positions, except for the radiation protection manager. The radiation protection manager shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 1, September 1975.

JUSTIFICATION FOR DEVIATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal. ANSI/ANS-3.1-1978 and Regulatory Guide 1.8, Revision 1, September 1975 are documents acceptable to the NRC staff.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.3, UNIT STAFF QUALIFICATIONS

There are no specific No Significant Hazards Considerations for this Section.

ATTACHMENT 4

ITS Section 5.4, Procedures

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

6.0 ADMINISTRATIVE CONTROLS

5.4 **6.8 PROCEDURES ~~AND PROGRAMS~~**

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a
5.4.1.b

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978~~,~~. ~~and those~~ required for implementing the requirements of NUREG 0737.

b. The emergency operating procedures and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33

A02

~~b. Refueling operations.~~

~~c. Surveillance and test activities of safety-related equipment.~~

A03

~~d. Not Used.~~

~~e. Not Used.~~

~~f. Not Used.~~

~~g. PROCESS CONTROL PROGRAM implementation.~~

LA01

~~h. OFFSITE DOSE CALCULATION MANUAL implementation.~~

A04

assurance

5.4.1.c i. Quality ~~Control Program~~ for effluent monitoring, ~~using the guidance in Regulatory Guide 1.21, Revision 1, June 1974~~.

assurance

LA02

5.4.1.c j. Quality ~~Control Program~~ for environmental monitoring ~~using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.~~

~~6.8.2 DELETED~~

Insert proposed Specification 5.4.1.d

M01

ADMINISTRATIVE CONTROLS

~~6.6 DELETED~~

~~6.7 DELETED~~

5.4 **6.8 PROCEDURES ~~AND PROGRAMS~~**

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a
5.4.1.b a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978~~,~~. ~~and those~~ required for implementing the requirements of NUREG 0737.

A02

b. The emergency operating procedures and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33

~~b. Refueling operations.~~

~~c. Surveillance and test activities of safety-related equipment.~~

A03

~~d. Not Used.~~

~~e. Not Used.~~

6.0 ADMINISTRATIVE CONTROLS

5.4 **6.8 PROCEDURES ~~AND PROGRAMS~~** (Continued)

~~f. Not Used.~~

~~g. PROCESS CONTROL PROGRAM implementation.~~

LA01

~~h. OFFSITE DOSE CALCULATION MANUAL implementation.~~

A04

assurance

5.4.1.c i. Quality ~~Control Program~~ for effluent monitoring~~, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974.~~

assurance

LA02

5.4.1.c j. Quality ~~Control Program~~ for environmental monitoring ~~using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.~~

Insert proposed Specification 5.4.1.d

M01

~~6.8.2 DELETED~~

~~6.8.3 DELETED~~

DISCUSSION OF CHANGES
ITS 5.4, PROCEDURES

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 6.8.1.a requires written procedures to be established, implemented, and maintained, in part, for those required for implementing the requirements of NUREG-0737. ITS 5.4.1.b provides a similar requirement specifically stating, "The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33." Supplement 1 to NUREG-0737 enclosed with Generic Letter 82-33 is part of the implementation of the Three Mile Island (TMI) action items of NUREG-0737. The purpose of the generic letter was to provide additional clarification regarding, in part, upgrade of emergency operating procedures associated with TMI Action Item I.C.1. As a result, the change clarifies the existing requirement to comply with NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33, for emergency operating procedures. This change is designated as an administrative change and is acceptable because it does not result in a technical change to the CTS.

A03 CTS 6.8.1.b and c require written procedures be established, implemented, and maintained covering refueling operations and surveillance and test activities of safety-related equipment, respectively. ITS 5.4.1 does not explicitly require these activities but rather requires written procedures shall be established, implemented, and maintained to the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 consistent with CTS 6.8.1.a. This changes the CTS by removing the specific wording of CTS 6.8.1.b and CTS 6.8.1.c. This change is considered administrative because the recommendations of Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 require procedures for refueling operations and surveillance tests for safety related activities. Therefore, specifically citing these requirements individually is unnecessary. This change is designated as administrative because it does not result in a technical change to the CTS.

A04 CTS 6.8.1.h requires written procedures be established, implemented, and maintained for OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation. ITS 5.4.1 requires procedures for various activities but does not specifically list the ODCM. This changes the CTS by removing the specific requirement for written procedures to implement the ODCM. This change is acceptable because implementing procedures for the ODCM are required by ITS 5.4.1.d. ITS 5.4.1.d (as described in DOC M01) requires that written procedures be established, implemented, and maintained for programs and manuals listed in ITS Section 5.5. ITS Section 5.5 includes the ODCM. Therefore, it is not necessary to specifically identify the ODCM in ITS 5.4.1. This

change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M01 ITS 5.4.1.d requires written procedures be established, implemented, and maintained for programs specified in Specification 5.5. The CTS does not include this requirement for any program except the OFFSITE DOSE CALCULATION MANUAL. This changes the CTS by adopting a new requirement for procedures to address programs described in ITS Section 5.5. The purpose of ITS 5.4.1.d is to ensure that written procedures are established, implemented, and maintained covering programs specified in ITS Section 5.5. This change is acceptable because it requires written procedures, including proper procedure control, to address programs required by ITS Section 5.5. This change is designated as more restrictive because it imposes new administrative requirements for procedures within the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, or ISI Program*) CTS 6.8.1.g requires that written procedures for the PROCESS CONTROL PROGRAM (PCP) be established, implemented, and maintained. The ITS does not include these requirements. This changes the CTS by moving the requirements to the Updated Safety Analysis Report (UFSAR). The removal of these details, which are related to meeting Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety.

The PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71 and written procedures are necessary to ensure compliance with regulations. Regulations provide an adequate level of control for the affected requirements, and thus, inclusion of this requirement in the Technical Specifications is not necessary. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LA02 (*Type 4 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, or ISI Program*) CTS 6.8.1.i and j require written procedures be established, implemented, and maintained

covering the Quality Control Program for effluent and environmental monitoring, respectively, "using the guidance in Regulatory Guide 1.21, Revision 1, 1974, and Regulatory Guide 4.1, Revision 1, April 1975." ITS 5.4.1.c does not include the Regulatory Guide references. This changes the CTS by moving the references to the regulatory guides to the UFSAR. The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for written procedures covering quality assurance for effluent and environmental monitoring. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because references for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

5.0 ADMINISTRATIVE CONTROLS

6.8 ## 5.4 Procedures

6.8.1 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

6.8.1.a a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978,

6.8.1.a b. The emergency operating procedures required to implement the requirements of NUREG-0737 and ~~to~~ NUREG-0737, Supplement 1, as stated in ~~[~~Generic Letter 82-33~~]~~, (2) (3)

6.8.1.i 6.8.1.j c. Quality assurance for effluent and environmental monitoring,

d. ~~Fire Protection Program implementation,~~ and (4)

DOC M01 ~~e.~~ All programs specified in Specification 5.5.

~~[f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.~~ (3)

~~Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.]~~

CTS

5.0 ADMINISTRATIVE CONTROLS

6.8 ## 5.4 Procedures

6.8.1 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

6.8.1.a a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978,

6.8.1.a b. The emergency operating procedures required to implement the requirements of NUREG-0737 and ~~to~~ NUREG-0737, Supplement 1, as stated in ~~[~~Generic Letter 82-33~~]~~, (2) (3)

6.8.1.i 6.8.1.j c. Quality assurance for effluent and environmental monitoring,

d. ~~Fire Protection Program implementation,~~ and (4)

DOC M01 ~~e.~~ All programs specified in Specification 5.5.

~~[f. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.~~ (3)

~~Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.]~~

JUSTIFICATION FOR DEVIATIONS
ITS 5.4, PROCEDURES

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Grammatical error corrected.

3. The ISTS contains bracketed information and/or values that are generic to Combustion Engineering vintage plants. The brackets are removed and the proper plant specific information/value is inserted to reflect the current licensing basis.

4. ISTS 5.4.1.d is deleted consistent with St. Lucie (PSL) Unit 1 and Unit 2 current licensing basis. PSL Units 1 and 2 transitioned to a risk-informed, performance-based fire protection program in accordance with National Fire Protection Association Standard NFPA 805 pursuant to 10 CFR 50.48(c). This transition was approved March 31, 2016 as issued in License Amendments 231 and 181, for Unit 1 and Unit 2 respectively (ADAMS Accession No. ML15344A346). As stated in the NRC Safety Evaluation supporting the license amendments, the administrative requirement that procedures be established, implemented, and maintained for Fire Protection Program implementation is contained in regulations 10 CFR 50.48(a) and 10 CFR 50.48(c); and in NFPA 805 Chapter 3. The NRC staff concluded that maintaining a procedure requirement for Fire Protection Program (FPP) implementation in Technical Specifications is redundant to the NFPA 805 requirement to establish FPP procedures, and as such, is unnecessary. Therefore, ISTS 5.4.1.d is not included in the ITS. The list in ITS 5.4.1 has been relabeled, as applicable.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.4, PROCEDURES

There are no specific No Significant Hazards Considerations for this Section.

ATTACHMENT 5

ITS Section 5.5, Programs and Manuals

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

6.0 ADMINISTRATIVE CONTROLS

~~6.8.3 DELETED~~

5.5 ~~6.8.4~~ The following programs shall be established, implemented, and maintained.

5.5.2 a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include ~~the Shutdown Cooling System~~, High Pressure Safety Injection ~~System~~, Containment Spray ~~System~~, and RCS Sampling. The program shall include the following:

A02

Low Pressure Safety Injection

5.5.2.a (i) Preventive maintenance and periodic visual inspection requirements, and

5.5.2.b (ii) Integrated leak test requirements for each system ~~at least once per 18 months~~.

LA01

in accordance with the Surveillance Frequency Control Program

SR 3.0.2

The provisions of ~~Specification 4.0.2~~ are applicable.

~~b. In-Plant Radioiodine Monitoring~~

~~A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~

LA02

~~(i) Training of personnel,~~

~~(ii) Procedures for monitoring, and~~

~~(iii) Provisions for maintenance of sampling and analysis equipment.~~

5.5.7 c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

5.5.7.a (i) Identification of a sampling schedule for the critical variables and control points of these variables,

5.5.7.b (ii) Identification of the procedures used to measure the values of the critical variables,

5.5.7.c (iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

5.5.7.d (iv) Procedures for the recording and management of data,

5.5.7.e (v) Procedures defining corrective actions for all off-control point chemistry conditions, and

5.5.7.f (vi) A procedure identifying ~~(a)~~ the authority responsible for the interpretation of the data, and ~~(b)~~ the sequence and timing of administrative events required to initiate corrective action.

~~d. Backup Method for Determining Subcooling Margin~~

~~A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:~~

~~(i) Training of personnel, and~~

~~(ii) Procedures for monitoring.~~

LA03

~~e. DELETED~~

5.5.3 f. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program ~~(1)~~ shall be contained in the ODCM, ~~(2)~~ shall be implemented by ~~operating~~ procedures, and ~~(3)~~ shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

functional capability

5.5.3.a 1) Limitations on the ~~operability~~ of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

5.5.3.b 2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten times the concentration values in 10 CFR 20.1001 - 20.240~~1~~2, Appendix B, Table 2, Column 2.

A03

5.5.3.c 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

5.5.3.d 4) Limitations on the annual and quarterly doses or dose commitment on a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5.5.3.e 5) Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

5.5.3.f 6) Limitations on the ~~operability~~ functional capability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

5.5.3.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at or beyond the SITE BOUNDARY shall be limited to the following:

a) For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b) For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ;

5.5.3.h 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.3.i 9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.3.j 10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

The provisions of ~~Specifications 4.0.2~~ SR 3.0.2 and ~~4.0.3~~ SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

~~g. Radiological Environmental Monitoring Program~~

~~A program shall be provided to monitor the radiation and radio-nuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of the environmental exposure pathways. The program shall (1) be contained in the ODCM;~~

LA04

~~(2) conform to the guidance of Appendix I to 10 CFR Part 50, and~~
~~(3) include the following:~~

~~1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.~~

~~2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

LA04

5.5.13 h. Containment Leakage Rate Testing Program

5.5.13.a A program to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program is in accordance with the guidelines contained in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J~~,~~." ~~except that the next Type A test performed after the December 8, 2005 Type A test shall be performed no later than December 8, 2020~~.

A04

5.5.13.b The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 42.~~8~~ psig. The containment design pressure is 44 psig.

77

M04 | R1

5.5.13.c The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

5.5.13.d Leakage rate acceptance criteria are:

5.5.13.d.1 a. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests, $\leq$ 0.75 L_a for Type A tests, and $\leq$ 0.096 L_a for secondary containment bypass leakage paths.

5.5.13.d.2 b. Air lock testing acceptance criteria are:

See ITS 3.6.3

5.5.13.d.2.a) 1) Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a.

5.5.13.d.2.b) 2) For ~~the~~ personnel air lock door ~~seal~~, leakage rate is < 0.01 L_a when pressurized to $\geq$ 1.0 P_a.

each

5.5.13.d.2.c) 3) For ~~the~~ emergency air lock door ~~seal~~, leakage rate is < 0.01 L_a when pressurized to $\geq$ 10 psig.

ADMINISTRATIVE CONTROLS (continued)

~~The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.~~ A05

5.5.13.e The provisions of ~~T.S. 4.0.3~~ are applicable to the Containment Leak Rate Testing Program.

SR 3.0.3

~~i. Deleted~~

Insert proposed 5.5.13.f A05

5.5.11 j. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

5.5.11.a 1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

5.5.11.b 2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

5.5.11.b.1 a. a change in the TS incorporated in the license; or

5.5.11.b.2 b. a change to the ~~updated~~ UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5.11.c 3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

5.5.11.b

5.5.11.d 4. Proposed changes that meet the criteria of ~~Specification 6.8.4.j.2.a or 6.8.4.j.2.b~~, above, shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

ADMINISTRATIVE CONTROLS (continued)

5.5.8 k. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems ~~at the frequencies specified in Regulatory Guide 1.52, Revision 3.~~ Insert 1 A06

5.5.8.a 1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ~~less than the value specified below~~ ≤ 0.05% when tested in accordance with ASME N510-1989 at the system flowrate specified below. ± 10%

ESF Ventilation System	Penetration	Flowrate
Control Room Emergency Ventilation System (CREVS)	~~≤ 0.05%~~	2000 ~~± 200 cfm~~
Shield Building Ventilation System SBVS	~~≤ 0.05%~~	6000 ~~± 600 cfm~~
Emergency Core Cooling System (ECCS) Area Ventilation System	~~≤ 0.05%~~	30,000 ~~± 3000 cfm~~

5.5.8.b 2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ~~less than the value specified below~~ when tested in accordance with ASME N510-1989 at the system flowrate specified below. ± 10%

ESF Ventilation System	Penetration	Flowrate
CREVS ~~Control Room Emergency Ventilation~~	≤ 0.05%	2000 ~~± 200 cfm~~
SBVS ~~Shield Building Ventilation System~~	~~≤ 0.05%~~	6000 ~~± 600 cfm~~
ECCS Area Ventilation System	~~≤ 0.05%~~	30,000 ~~± 3000 cfm~~

5.5.8.c 3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 3, shows the methyl iodide penetration ~~less than the value specified below~~ when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and ~~the~~ a relative humidity ~~specified below~~ ≤ 70%.

ESF Ventilation System	Penetration	RH
CREVS ~~Control Room Emergency Ventilation~~	≤ 2.5%	~~70%~~
SBVS ~~Shield Building Ventilation System~~	~~≤ 2.5%~~	~~70%~~
ECCS Area Ventilation System	~~≤ 2.5%~~	~~70%~~

5.5.8.d 4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below. ± 10%

ESF Ventilation System	Delta P	Flowrate
CREVS ~~Control Room Emergency Ventilation~~	< 4.15" W.G.	2000 ~~± 200 cfm~~
SBVS ~~Shield Building Ventilation System~~	≤ 6.15" W.G.	6000 ~~± 600 cfm~~
ECCS Area Ventilation System	< 4.15" W.G.	30,000 ~~± 3000 cfm~~

At a frequency in accordance with the Surveillance Frequency Control Program

5.5.8.f 5. ~~At least once per 18 months~~, demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989. LA01

ESF Ventilation System	Wattage
SBVS ~~Shield Building Ventilation System~~	
Main Heaters	30 ± 3 kW
Auxiliary Heaters	1.5 ± 0.25 kW

ITS

Insert 1

5.5.8 and the Fuel Pool Area Ventilation System in accordance with Regulatory Guide 1.52, Revision 3, ASME N510-1989, ASTM D3803-1989 and, ANSI N510-1975, as described herein.

ADMINISTRATIVE CONTROLS (continued)

M01

The provisions of SR ~~4.0.2~~ and SR ~~4.0.3~~ are applicable to the VFTP test frequencies.

3.0.2 3.0.3

5.5.6 l. Steam Generator (SG) Program

Insert proposed 5.5.5, "RCP Flywheel Inspection Program"

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

5.5.6.a a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

5.5.6.b b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gpm total through all SGs and 0.25 gpm through any one SG.

3. The operational leakage performance criterion is specified in LCO ~~3.4.6.2.c~~, "Reactor Coolant System Operational Leakage."

3.4.13

plugging

5.5.6.c c. Provisions for SG tube ~~repair~~ criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

A07

ADMINISTRATIVE CONTROLS (continued)

5.5.6 l. Steam Generator (SG) Program (continued)

5.5.6.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube ~~repair~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tube may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

plugging

A07

5.5.6.d.1 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

5.5.6.d.2 2. Inspect 100% of the tubes at ~~sequential periods of 144, 108, 72, and, thereafter, 60~~ 96 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. ~~In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.~~

L01

affected and potentially affected

5.5.6.d.3 3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall ~~not exceed 24 effective full power months or one~~ be at the next refueling outage ~~(whichever is less)~~. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

5.5.6.e e. Provisions for monitoring operational primary-to-secondary leakage.

M02

Insert proposed 5.5.14, "Battery Monitoring and Maintenance Program"

5.5.15 m. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

ADMINISTRATIVE CONTROLS (continued)

5.5.15 m. Control Room Envelope Habitability Program (continued)

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency ~~of 36 months on a STAGGERED TEST BASIS~~. The results shall be trended and used as part of the 36 month assessment of the CRE boundary.

in accordance with the Surveillance Frequency Control Program

LA01

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

3.0.2

f. The provisions of SR ~~4.0.2~~ are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

n. Diesel Fuel Oil Testing Program

5.5.10

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

5.5.10.a (i) Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,
2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
3. A clear and bright appearance with proper color or a water and sediment content within limits;

5.5.10.b (ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following ~~sampling and~~ addition to storage tanks; and

5.5.10.c (iii) Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested ~~every 31 days~~.

LA01

in accordance with the Surveillance Frequency Control Program

The provisions of SR ~~4.0.2~~ and SR ~~4.0.3~~ are applicable to the Diesel Fuel Oil Testing Program test frequencies.

3.0.2

3.0.3

5.5.16 o. Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of frequencies of those Surveillance Requirements for which the frequency is controlled by the program.

b. Changes to the frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

SR 3.0.2 SR 3.0.3

c. The provisions of ~~Surveillance Requirements~~ 4.0.2 and ~~4.0.3~~ are applicable to the frequencies established in the Surveillance Frequency Control Program.

~~p. Snubber Testing Program~~ R1

~~This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:~~

~~1. This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.~~

~~2. The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to the conditions listed in 10 CFR 50.55a(b) and subject to Commission approval.~~ LA08

~~3. The program shall, as required by 10 CFR 50.55a(b)(3)(v), meet Subsection ISTA, "General Requirements" and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants".~~

~~4. The 120-month program updates shall be made in accordance with 10 CFR 50.55a(g)(4), 10 CFR 50.55a(g)(3)(v) and 10 CFR 50.55a(b) (including 10 CFR 50.55a(b)(3)(v)) subject to the conditions listed therein.~~

5.5.4 q. Component Cyclic or Transient Limit Program U

The program provides controls to track the FSAR, Section 5.2, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.17 r. Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, ~~"Risk-Informed Technical Specifications Initiative 4b:~~ Risk-Managed Technical Specifications (RMTS) Guidelines~~,~~." Revision 0 ~~-A, November 2006.~~ The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODES 1 and 2;

5.5.17

c. When a RICT is being used, any plant configuration change within the scope of the Configuration Risk Management Program must be considered for the effect on the RICT.

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required If the plant configuration change would lower plant risk and would result in a longer RICT.

d. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

e. If the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

1. Numerically accounting for the increased possibility of CCF in the RICT calculation, or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

s. Safety Function Determination Program (SFDP)

5.5.12

limitations and remedial or compensatory

A08

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system ~~ACTIONS~~. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

Conditions and Required Actions

A09

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

b Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists,

Completion Time

A09

c. Provisions to ensure that an inoperable supported system's ~~allowed outage time~~ is not inappropriately extended as a result of multiple support system inoperabilites, and

d. Other appropriate limitations and remedial or compensatory actions.

5.5.12 A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

Conditions and Required Actions

A09

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate ~~ACTIONS~~ of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate ~~ACTIONS~~ to enter are those of the support system.

Conditions and Required Actions

A09

6.9 REPORTING REQUIREMENTS

See ITS 5.6

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following:

(1) receipt of an operating license,

(2) amendment of the license involving a planned increase in power level,

(3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and

(4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

1. Shall be documented and this documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after the approval of the plant manager.

See CTS 6.0

5.5.1 ## 6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

and records of reviews performed shall be retained.

M03

5.5.1.a 1. Shall be documented ~~and~~ this documentation shall contain:

5.5.1.a.1 a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

5.5.1.a.2 b. A determination that the change ~~will~~ maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

5.5.1.b 2. Shall become effective after the approval of the plant manager.

5.5.1.c 3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the ~~Annual~~ Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

DEFINITIONS

IDENTIFED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system (Primary-to-secondary leakage).

INSERVICE TESTING PROGRAM

1.16 The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

MEMBER(S) OF THE PUBLIC

1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.a
5.5.1.b

~~1.18~~ THE OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain ~~(1)~~ the Radioactive Effluent Controls and Radiological Environmental Monitoring ~~Programs required by Section 6.8.4~~ and ~~(2)~~ descriptions of the information that should be included in the Annual Radiological Environmental Operating and ~~Annual~~ Radioactive Effluent Release Reports required by Specifications ~~6.9.1.7~~ and ~~6.9.1.8~~.

activities

5.6.1

5.6.2

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM

See ITS 3.6.9

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent shield building ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one shield building ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE

Action not applicable when second shield building ventilation system intentionally made inoperable.

b. With two shield building ventilation systems inoperable, within 1 hour verify at least one train of containment spray is OPERABLE, and restore at least one shield building ventilation system to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each shield building ventilation system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 continuous minutes with the heaters on.

b. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

5.5.8

c. In accordance with the Surveillance Frequency Control Program by:

5.5.8.e

1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.

2. Verifying that the filtration system starts automatically on a Containment Isolation Signal (CIS).

See ITS 3.6.9

3. Verifying that the filter cooling makeup air and cross connection valves can be manually opened.

See ITS 3.6.7

4. Verifying that each system produces a negative pressure of $\geq$ 2.0 inches W.G. in the annulus within 2 minutes after a Containment Isolation Signal (CIS).

PLANT SYSTEMS

See ITS 3.7.12

3/4.7.8 ECCS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two independent ECCS area exhaust air filter trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one ECCS area exhaust air filter train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each ECCS area exhaust air filter train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. By performing required ECCS area ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

5.5.8

c. In accordance with the Surveillance Frequency Control Program:

5.5.8.e

1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.

See ITS 3.7.12

2. Verifying that the filter train starts on a Safety Injection Actuation Signal.

ITS

See ITS 3.8.6

A01

ITS 5.5

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENT

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts	> 2.07 volts
Specific Gravity(a)	≥ 1.195(b)	≥ 1.190 Average of all connected cells > 1.200	Not more than .020 below the average of all connected cells Average of all connected cells ≥ 1.190(b)

5.5.14. b.1 and b.2

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amps when on charge.

5.5.14

~~(c) Corrected for average electrolyte temperature.~~

LA05

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within the limits within 7 days.

(3) With any Category B parameter not within its allowable value, declare the battery inoperable.

See ITS 3.8.6

REFUELING OPERATIONS

FUEL POOL VENTILATION SYSTEM – FUEL STORAGE

See ITS 3.7.17

LIMITING CONDITION FOR OPERATION

3.9.12 At least one fuel pool ventilation system shall be OPERABLE.

APPLICABILITY: Whenever recently irradiated fuel is in the spent fuel pool.

ACTION:

a. With no fuel pool ventilation system OPERABLE, suspend all operations involving movement of recently irradiated fuel within the spent fuel pool or crane operation with loads over the recently irradiated spent fuel until at least one fuel pool ventilation system is restored to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel pool ventilation system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

5.5.8 Insert 2

b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.8.h

1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

5.5.8.g

2. Verifying that the HEPA filter banks remove ≥ 99% of the ~~DOP~~ dioctyl phthalate when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

5.5.8.i

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of ≥ 85% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989 (30°C, 95% RH). The carbon samples not obtained from test canisters shall be prepared by either:

5.5.8.i.1

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

5.5.8.i.2

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

~~4. Verifying a system flow rate of 10,350 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.~~

A10

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

5.5.8

c. In accordance with the Surveillance Frequency Control Program by:

5.5.8.j

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4.15 inches Water Gauge while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

5.5.8.k

2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

3. Verifying that the ventilation system maintains the spent fuel storage pool area at a negative pressure of ≥ 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

See ITS 3.7.17

5.5.8

d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the ~~DOP~~ dioctyl phthalate when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

5.5.8.g

5.5.8

e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

5.5.8.h

~~Pages 3/4 11-2 through 3/4 11-13 (Amendment No. 123) have been deleted from the Technical Specifications. The next page is 3/4 11-14.~~

A01

~~RADIOACTIVE EFFLUENTS~~

and Storage Tank Radioactivity Monitoring Program

EXPLOSIVE GAS ~~MIXTURE~~

Add proposed ITS 5.5.9 generic program statement

A11

~~LIMITING CONDITION FOR OPERATION~~

5.5.9.a

3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited ~~to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.~~

~~APPLICABILITY:~~ ~~At all times.~~

LA06

~~ACTION:~~

~~a. With the concentration of oxygen in the waste gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.~~

~~b. With the concentration of oxygen in the waste gas decay tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and immediately commence reduction of the concentration of oxygen to less than or equal to 2% by volume.~~

~~c. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

5.5.9 a and b

4.11.2.5.1 The concentration of oxygen in the waste gas decay tank shall be determined to be within the above limits by ~~continuously*~~ monitoring the waste gases in the ~~on service~~ waste gas decay tank.

5.5.9 a and b

4.11.2.5.2 ~~With the oxygen concentration in the on service waste gas decay tank greater than 2% by volume as determined by Specification 4.11.2.5.1,~~ the concentration of hydrogen in the waste gas decay tank shall be determined to be within the above limits ~~by gas partitioner sample at least once per 24 hours.~~

LA06

~~* When continuous monitoring capability is inoperable, waste gases shall be monitored in accordance with the actions specified for the Waste Gas Decay Tanks Explosive Gas Monitoring System in Chapter 13 of the Updated Final Safety Analysis Report.~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

A11

A01

~~RADIOACTIVE EFFLUENTS~~

~~GAS STORAGE TANKS~~

Explosive Gas and Storage Tank Radioactivity Monitoring Program

Add proposed ITS 5.5.9 generic program statement

A11

~~LIMITING CONDITION FOR OPERATION~~

5.5.9.a

3.11.2.6 The quantity of radioactivity contained in each gas storage tank ~~shall be limited to less than or equal to 165,000 curies noble gases (considered as Xe-133).~~

LA06

~~**APPLICABILITY:** At all times.~~

~~**ACTION:**~~

~~a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

5.5.9 a and b

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit ~~in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank when reactor coolant system activity exceeds 518.9 µCi/gram DOSE EQUIVALENT XE-133.~~

LA06

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

A11

6.0 ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

5.5 **~~6.8.4~~** The following programs shall be established, implemented, and maintained.

5.5.2 a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include ~~the Shutdown Cooling System~~, High Pressure Safety Injection ~~System~~, Containment Spray System, and RCS Sampling. The program shall include the following:

A02

Low Pressure Safety Injection

5.5.2.a (i) Preventive maintenance and periodic visual inspection requirements, and

5.5.2.b (ii) Integrated leak test requirements for each system ~~at least once per 18 months~~.

LA01

in accordance with the Surveillance Frequency Control Program

SR 3.0.2

The provisions of ~~Specification 4.0.2~~ are applicable.

~~b. In-Plant Radioiodine Monitoring~~

~~A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~

LA02

~~(i) Training of personnel,~~

~~(ii) Procedures for monitoring, and~~

~~(iii) Provisions for maintenance of sampling and analysis equipment.~~

PROCEDURES AND PROGRAMS (continued)

5.5.7

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

5.5.7.a

(i) Identification of a sampling schedule for the critical variables and control points of these variables,

5.5.7.b

(ii) Identification of the procedures used to measure the values of the critical variables,

5.5.7.c

(iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

5.5.7.d

(iv) Procedures for the recording and management of data,

5.5.7.e

(v) Procedures defining corrective actions for all off-control point chemistry conditions, and

5.5.7.f

(vi) A procedure identifying ~~(a)~~ the authority responsible for the interpretation of the data, and ~~(b)~~ the sequence and timing of administrative events required to initiate corrective action.

~~d. Backup Method for Determining Subcooling Margin~~

~~A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:~~

~~(i) Training of personnel, and~~

~~(ii) Procedures for monitoring.~~

LA03

~~e. DELETED~~

5.5.3 f. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program ~~(1)~~ shall be contained in the ODCM, ~~(2)~~ shall be implemented by operating procedures, and ~~(3)~~ shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

functional capability

5.5.3.a 1) Limitations on the ~~operability~~ of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

5.5.3.b 2) Limitations on the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten times the concentration values in 10 CFR 20.1001 – 20.2401, Appendix B, Table 2, Column 2.

2

A03

5.5.3.c 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

5.5.3.d 4) Limitations on the annual and quarterly doses or dose commitment on a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5.5.3.e 5) Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

functional capability

5.5.3.f 6) Limitations on the ~~operability~~ and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

5.5.3.g 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at or beyond the SITE BOUNDARY shall be limited to the following:

a) For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b) For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ;

5.5.3.h 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.3.i 9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater

than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

5.5.3.j

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

SR 3.0.2 SR 3.0.3

The provisions of ~~Specifications 4.0.2~~ and ~~4.0.3~~ are applicable to the Radioactive Effluent Controls Program surveillance frequency.

~~g. Radiological Environmental Monitoring Program~~

LA04

~~A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of the environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:~~

~~1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.~~

~~2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

5.5.13

h. Containment Leakage Rate Testing Program

5.5.13.a

A program to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program is in accordance with the guidelines contained in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 18, 2007 Type A test shall be performed no later than December 18, 2022.

5.5.13.b

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 43.4~~8~~ 3 psig. The containment design pressure is 44 psig.

M04 | R1

5.5.13.c

The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

ADMINISTRATIVE CONTROLS (Continued)

5.5.13.d Leakage rate acceptance criteria are:

5.5.13.d.1 a. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests, $\leq$ 0.75 L_a for Type A tests, and $\leq$ 0.096 L_a for secondary containment bypass leakage paths.

See ITS 3.6.3

5.5.13.d.2 b. Air lock testing acceptance criteria are:

5.5.13.d.2.a) 1) Overall air lock leakage is $\leq$ 0.05 L_a when tested at $\geq$ P_a.

air lock

5.5.13.d.2.b) 2) For each door ~~seal~~, leakage rate is < 0.01 L_a when pressurized to $\geq$ P_a.

~~The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.~~

A05

5.5.13.e The provisions for ~~T.S. 4.0.3~~ are applicable to the Containment Leak Rate Testing Program.

SR 3.0.3

~~i. Deleted~~

Insert proposed 5.5.13.f

A05

ADMINISTRATIVE CONTROLS (continued)

5.5.11 j. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

5.5.11.a 1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

5.5.11.b 2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

5.5.11.b.1 a. a change in the TS incorporated in the license; or

5.5.11.b.2 b. a change to the ~~updated~~ UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.5.11.c 3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

5.5.11.b

5.5.11.d 4. Proposed changes that meet the criteria of ~~Specification 6.8.4.j.2.a or 6.8.4.j.2.b,~~ above, shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.8 k. Ventilation Filter Testing Program (VFTP)

in accordance with

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems ~~at the frequencies specified in~~ Regulatory Guide 1.52, Revision 3~~.~~ , ASME N510-1989 and ASTM D3803-1989, as described herein.

A06

5.5.8.a 1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ~~less than the value specified below~~ when tested in accordance with ASME N510-1989 at the system flowrate specified below.

≤ 0.05%

± 10%

ESF Ventilation System	Penetration	Flowrate
Control Room Emergency Air Cleanup	~~≤ 0.05%~~	2000 ~~± 200 cfm~~
Shield Building Ventilation System	~~≤ 0.05%~~	6000 ~~± 600 cfm~~
~~ECCS~~ Area Ventilation System	~~≤ 0.05%~~	30,000 ~~± 3000 cfm~~

System (CREVS)

SBVS

Emergency Core Cooling System (ECCS)

5.5.8.b 2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ~~less than the value specified~~ below when tested in accordance with ASME N510-1989 at the system flowrate specified below.

≤ 0.05%

± 10%

ESF Ventilation System	Penetration	Flowrate
~~Control Room Emergency Air Cleanup~~	~~≤ 0.05%~~	2000 ~~± 200 cfm~~
~~Shield Building Ventilation System~~	~~≤ 0.05%~~	6000 ~~± 600 cfm~~
ECCS Area Ventilation System	~~≤ 0.05%~~	30,000 ~~± 3000 cfm~~

CREVS

SBVS

ADMINISTRATIVE CONTROLS (continued)

k. Ventilation Filter Testing Program (VFTP) (continued)

5.5.8.c

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 3, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and ~~the~~ relative humidity ~~specified below~~. ≤ 95% a

ESF Ventilation System	Penetration	~~RH~~
CREVS → ~~Control Room Emergency Air Cleanup~~	≤ 0.175%	~~95%~~
SBVS → ~~Shield Building Ventilation System~~	≤ 2.5%	~~95%~~
ECCS Area Ventilation System	≤ 2.5%	~~95%~~

5.5.8.d

4. For the ~~Control Room Emergency Air Cleanup System~~ (CREVS) and the ECCS Area Ventilation System, demonstrate that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below. For the ~~Shield Building Ventilation System~~ (SBVS), demonstrate that the pressure drop across the combined demisters, electric heaters, HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
CREVS → ~~Control Room Emergency Air Cleanup~~	< 7.4" W.G.	2000 ~~± 200 cfm~~
SBVS → ~~Shield Building Ventilation System~~	< 8.5" W.G.	6000 ~~± 600 cfm~~
ECCS Area Ventilation System	< 4.35" W.G.	30,000 ~~± 3000 cfm~~

5.5.8.f

5. ~~At least once per 18 months~~ At a frequency in accordance with the Surveillance Frequency Control Program, demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989. LA01

ESF Ventilation System	Wattage
SBVS → ~~Shield Building Ventilation System~~	
Main Heaters	30 ± 3 kW
Auxiliary Heaters	1.5 ± 0.25 kW

The provisions of SR ~~4~~.0.2 and SR ~~4~~.0.3 are applicable to the VFTP test frequencies. 3

5.5.6

l. Steam Generator (SG) Program

1. A SG Program shall be established and implemented for the replacement SGs to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following provisions:

5.5.6.a

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

ADMINISTRATIVE CONTROLS (continued)

l. Steam Generator (SG) Program (continued)

5.5.6

1. (continued)

5.5.6.b

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.5 gallons per minute total through all SGs and 0.25 gallons per minute through any one SG.

3. The operational leakage performance criterion is specified in LCO ~~3.4.6.2.c~~, "Reactor Coolant System Operational Leakage."

3.4.13

ADMINISTRATIVE CONTROLS (continued)

5.5.6 l. Steam Generator (SG) Program (continued)

1. (continued)

plugging

A07

5.5.6.c c. Provisions for SG tube ~~repair~~ criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

5.5.6.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube ~~repair~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

plugging

A07

5.5.6.d.1 1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

5.5.6.d.2 2. Inspect 100% of the tubes at ~~sequential periods of 144, 108, 72, and, thereafter, 60~~ 96 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. ~~In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.~~

L01

affected and potentially affected

5.5.6.d.3 3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall ~~not exceed 24 effective full power months or one~~ be at the next refueling outage ~~(whichever is less)~~. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

5.5.6.e e. Provisions for monitoring operational primary-to-secondary leakage

Insert proposed 5.5.14, "Battery Monitoring and Maintenance Program"

M02

~~ADMINISTRATIVE CONTROLS (continued)~~

~~PAGES 6-15g AND 6-15h HAVE BEEN DELETED.~~
~~THE NEXT PAGE IS 6-15i.~~

5.5.15

m. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Cleanup System (CREACS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACS, operating at the flow rate required by the VFTP, ~~at a Frequency of 36 months on a STAGGERED TEST BASIS~~ in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the 36 month assessment of the CRE boundary. LA01

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR ~~4~~3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.10 n. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

5.5.10.a (i) Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. A clear and bright appearance with proper color or a water and sediment content within limits;

in accordance with the Surveillance Frequency Control Program

5.5.10.b (ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following ~~sampling and~~ addition to storage tanks; and

LA01

5.5.10.c (iii) Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested ~~every 31 days~~.

3 3

The provisions of SR ~~4~~.0.2 and SR ~~4~~.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

Insert proposed requirements based on RCP flywheel material

L02

5.5.5 o. Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel by either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years.

~~p. Snubber Testing Program~~ R1

~~This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:~~

~~1. This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.~~

~~2. The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval.~~

LA08

~~3. The program shall, as required by 10 CFR 50.55a(b)(3)(v)(B), meet Subsection ISTA, "General Requirements" and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants".~~

~~4. The 120-month program updates shall be made in accordance with 10 CFR 50.55a(g)(4), 10 CFR 50.55a(g)(3)(v) and 10 CFR 50.55a(b) (including 10 CFR 50.55a(b)(3)(v)(B)) subject to the limitations and modifications listed therein.~~

ADMINISTRATIVE CONTROLS

5.5.16 q. Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of frequencies of those Surveillance Requirements for which the frequency is controlled by the program.

b. Changes to the frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

SR 3.0.2 SR 3.0.3

c. The provisions of ~~Surveillance Requirements 4.0.2~~ and ~~4.0.3~~ are applicable to the frequencies established in the Surveillance Frequency Control Program.

5.5.4 r. Component Cyclic or Transient Limit Program

U

The Program provides controls to track the FSAR, Section 3.9, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.17 s. Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, ~~"Risk-Informed Technical Specifications Initiative 4b:~~ Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0 ~~-A, November 2006~~. The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODES 1 and 2;

c. When a RICT is being used, any plant configuration change within the scope of the Configuration Risk Management Program must be considered for the effect on the RICT.

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

ADMINISTRATIVE CONTROLS

5.5.17 e. If the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

1. Numerically accounting for the increased possibility of CCF in the RICT calculation, or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

5.5.12 t. Safety Function Determination Program (SFDP)

limitations and remedial or compensatory

A08

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system ~~ACTIONS~~. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

Conditions and Required Actions

A09

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

b Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists,

Completion Time

A09

c. Provisions to ensure that an inoperable supported system's ~~allowed outage time~~ is not inappropriately extended as a result of multiple support system inoperabilites, and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

Conditions and Required Actions

A09

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate ~~ACTIONS~~ of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate ~~ACTIONS~~ to enter are those of the support system.

Conditions and Required Actions

A09

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

1. Shall be documented and this documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after the approval of the plant manager.

See CTS 6.0

5.5.1

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

and records of reviews performed shall be retained.

M03

5.5.1.a 1. Shall be documented ~~and~~ this documentation shall contain:

5.5.1.a.1 a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

5.5.1.a.2 b) A determination that the change ~~will~~ maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

5.5.1.b 2. Shall become effective after the approval of the plant manager.

5.5.1.c 3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the ~~Annual~~ Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

DEFINITIONS

INSERVICE TESTING PROGRAM

1.16 The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

MEMBER(S) OF THE PUBLIC

1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

5.5.1.a
5.5.1.b

~~1.18~~ THE OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain ~~(1)~~ the Radioactive Effluent Controls and Radiological Environmental Monitoring ~~Programs required by Section 6.8.4~~ and ~~(2)~~ descriptions of the information that should be included in the Annual Radiological Environmental Operating and ~~Annual~~ Radioactive Effluent Release Reports required by Specifications ~~6.9.1.7~~ and ~~6.9.1.8~~.

activities

5.6.1

5.6.2

OPERABLE – OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE – MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

See ITS 1.1

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM (SBVS)

See ITS 3.6.9

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Shield Building Ventilation Systems shall be OPERABLE.

APPLICABILITY: At all times in MODES 1, 2, 3, and 4.
In addition, during movement of recently irradiated fuel assemblies or during crane operations with loads over recently irradiated fuel assemblies in the Spent Fuel Storage Pool in MODES 5 and 6.

ACTION:

a. With the SBVS inoperable solely due to loss of the SBVS capability to provide design basis filtered air evacuation from the Spent Fuel Pool area, only ACTION-c is required. If the SBVS is inoperable for any other reason, concurrently implement ACTION-b and ACTION-c.

b. (1) With one SBVS inoperable in MODE 1, 2, 3, or 4, restore the inoperable system to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when second SBVS intentionally made inoperable.

(2) With both SBVSs inoperable, within 1 hour verify at least one train of containment spray is OPERABLE, and restore at least one SBVS to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. (1) With one SBVS inoperable in any MODE, restore the inoperable system to OPERABLE status within 7 days; otherwise, suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

(2) With both SBVS inoperable in any MODE, immediately suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each Shield Building Ventilation System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 continuous minutes with the heaters on.

5.5.8

In accordance with the frequencies of Regulatory Guide 1.52, Revision 3, except that the testing frequency of "24 months" is required at a frequency in

b. ~~In~~ accordance with the Surveillance Frequency Control Program ~~or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:~~

A12

~~1. Performing a visual examination of SBVS in accordance with ASME N510-1989.~~

LA07

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

5.5.8.e

2. Performing airflow distribution to HEPA filters and charcoal adsorbers in accordance with ASME N510-1989. The distribution shall be ± 20% of the average flow per unit.

c. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

d. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that the system starts on a Unit 2 containment isolation signal and on a fuel pool high radiation signal.

2. Verifying that the filter cooling makeup and cross connection valves can be manually opened.

3. Verifying that each system produces a negative pressure of greater than or equal to 2.0 inches WG in the annulus within 99 seconds after a start signal.

4. Verifying that each system achieves a negative pressure of greater than 0.125 inch WG in the fuel storage building after actuation of a fuel storage building high radiation test signal.

See ITS 3.6.9

See ITS 3.6.7

ITS

See ITS 3.8.6

A01

ITS 5.5

TABLE 4 8-2

BATTERY SURVEILLANCE REQUIREMENT

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	> Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(c)	> 2.07 volts
			Not more than .020 below the average of all connected cells
Specific Gravity(a)	≥ 1.195(b)	≥ 1.190	Average of all connected cells ≥ 1.190(b)
		Average of all connected cells > 1.200	

5.5.14. b.1 and b.2

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amps when on charge.

5.5.14 ~~(c) Corrected for average electrolyte temperature.~~

LA05

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within the limits within 7 days.

(3) With any Category B parameter not within its allowable value, declare the battery inoperable.

See ITS 3.8.6

~~Pages 3/4 11-2 through 3/4 11-13 (Amendment No. 61) have been deleted from the Technical Specifications. The next page is 3/4 11-14.~~

A01

~~RADIOACTIVE EFFLUENTS~~

and Storage Tank Radioactivity Monitoring Program

EXPLOSIVE GAS ~~MIXTURE~~

A11

Add proposed ITS 5.5.9 generic program statement

~~LIMITING CONDITION FOR OPERATION~~

5.5.9.a

3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited ~~to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.~~

~~**APPLICABILITY:** At all times.~~

LA06

~~**ACTION:**~~

~~a. With the concentration of oxygen in the waste gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.~~

~~b. With the concentration of oxygen in the waste gas decay tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and immediately commence reduction of the concentration of oxygen to less than or equal to 2% by volume.~~

~~c. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

5.5.9 a and b

4.11.2.5.1 The concentration of oxygen in the waste gas decay tank shall be determined to be within the above limits by ~~continuously*~~ monitoring the waste gases in the ~~on service~~ waste gas decay tank.

5.5.9 a and b

4.11.2.5.2 ~~With the oxygen concentration in the on service waste gas decay tank greater than 2% by volume as determined by Specification 4.11.2.5.1,~~ the concentration of hydrogen in the waste gas decay tank shall be determined to be within the above limits ~~by gas partitioner sample at least once per 24 hours.~~

LA06

~~* When continuous monitoring capability is inoperable, waste gases shall be monitored in accordance with the actions specified for the Waste Gas Decay Tanks Explosive Gas Monitoring System in Chapter 13 of the Updated Final Safety Analysis Report.~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

A11

~~RADIOACTIVE EFFLUENTS~~

~~GAS STORAGE TANKS~~

Explosive Gas and Storage Tank Radioactivity Monitoring Program

A11

Add proposed ITS 5.5.9 generic program statement

~~LIMITING CONDITION FOR OPERATION~~

5.5.9.a

3.11.2.6 The quantity of radioactivity contained in each gas storage tank ~~shall be limited to less than or equal to 165,000 curies noble gases (considered as Xe-133).~~

LA06

~~**APPLICABILITY:** At all times.~~

~~**ACTION:**~~

~~a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank.~~

~~b. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

5.5.9 a and b

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit ~~in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank when reactor coolant system activity exceeds 518.9 µCi/gram DOSE EQUIVALENT XE-133.~~

LA06

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

A11

DISCUSSION OF CHANGES
ITS 5.5, PROGRAMS AND MANUALS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 6.8.4.a contains a program requiring leakage testing of primary coolant sources outside containment. Included in the listing of systems is the Shutdown Cooling System. ITS 5.5.2 requires the same program and has renamed the Shutdown Cooling System to the Low Pressure Safety Injection System. This changes the CTS by renaming the low pressure cooling/injection system.

The purpose of this administrative program is to ensure systems outside containment that have the potential to contain highly radioactive liquid following a design basis accident (DBA) are leakage tested periodically. The Shutdown Cooling System performs the function of normal shutdown decay heat removal and low pressure safety injection. The titles Low Pressure Safety Injection System and Shutdown Cooling System may be used interchangeable when discussing the system. Since the purpose of the program is to ensure leak tight systems following a DBA, it is more appropriately to use the safety injection function nomenclature. This change is consistent with the ISTS. The change is designated as administrative and is acceptable because it does not result in technical changes to the CTS.

A03 CTS 6.8.4.f 2) requires the Radioactive Effluent Controls Program to provide limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas and references Appendix B of 10 CFR 20.1001 – 20.2401. ITS 5.5.3 contains the same requirement but references Appendix B of 10 CFR 20.1001 – 20.2402 consistent with the ISTS. This changes the CTS by changing the 10 CFR 20 reference from 2401 to 2402. |R1

This change is considered acceptable because the 10 CFR 20 regulation encompasses 1001 – 2402. The change is designated as an administrative change to correct the regulation reference and is acceptable because it does not result in technical change to the CTS programmatic requirement.

A04 **Unit 1 only:** CTS 6.8.4.h contains an exception to Containment Leakage Rate Testing Program requirements required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J in accordance with the guidelines contained in NEI-94-01. CTS 6.8.4.h states, in part, that the next Type A test performed after the December 8, 2005 Type A test shall be performed no later than December 8, 2020. ITS 5.5.13 does not include this Type A test exception. This changes the CTS by removing a 10 CFR 50, Appendix J exception that is no longer applicable.

The Type A test was performed in November 2019. Therefore, this exception to the Type A containment leakage rate testing frequency specified in the regulation and supporting documents is no longer required. The change is designated as administrative and is acceptable because it does not result in technical changes to the CTS.

A05 TS 6.8.4.h states, in part, that the provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program. ITS 5.5.13.f states, "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J." This changes the CTS by exchanging the requirement that CTS 4.0.2 does not apply with the statement to clarify that Containment Leakage Rate Testing Program test frequencies required by a regulation cannot be extended by the use of any Technical Specification allowance, including CTS 4.0.2.

The change is designated as an administrative change to clarify that no generic testing frequency allowance shall be construed to allow modifying a testing Frequency required pursuant to a regulation and is acceptable because it does not result in technical change to the CTS programmatic requirement.

A06 Unit 1 CTS 4.6.6.1.c.1 and 4.7.8.1.c.1 and Unit 2 CTS 4.6.6.1.b.2 require a test to verify airflow distribution to HEPA filters and charcoal adsorbers in accordance with ASME N510-1989 for the Shield Building Ventilation System (SBVS) and the Unit 1 Emergency Core Cooling System (ECCS) Area Ventilation System. Additionally, Unit 1 CTS 4.9.12 provides filter testing requirements for the Fuel Pool Area Ventilation System, which are tested in accordance with ANSI N510-1975 and ASTM D3803-1989, as applicable. ITS includes the Surveillances associated with the ventilation filter testing in the Ventilation Filter Testing Program (ITS 5.5.8). As such, the general program statement has been modified to include, in addition to Regulatory Guide 1.52, the ASME, ANSI, and ASTM Standards applicable to the filter testing and a listing of which Surveillance Frequency requirements are associated with each filter test. This changes the CTS by moving additional ventilation filter testing Surveillances associated with the SBVS, Unit 1 ECCS Area Ventilation System, and Unit 1 Fuel Pool Area Ventilation System to the Ventilation Filter Testing Program.

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A07 CTS 6.8.4 l.c, states the provisions for steam generator (SG) tube repair criteria. ITS 5.5.6.c refers to the SG tube plugging criteria. This changes the CTS by referring to SG plugging criteria instead of "repair" criteria.

SG repair applied to the original SGs. However, the unit contains replacement SGs containing Alloy 690TT Tubing. As further stated in CTS 6.8.4.l.c, if SG tube degradation is discovered, the tubes shall be plugged. Therefore, there is no repair criteria, only plugging criteria, defined in CTS. This change is designated as administrative and is acceptable because it does not result in technical changes to the CTS.

A08 Unit 1 CTS 6.8.4.s and Unit 2 CTS 6.8.4.t state that the Safety Function Determination Program (SFDP) ensures loss of safety function is detected and appropriate actions taken, and other appropriate actions may be taken as a result of the support system inoperability. ITS 5.5.12 states that other appropriate limitations and remedial or compensatory actions may be taken as a result of the support system inoperability. The changes the CTS by adding additional clarification on what other appropriate action may be taken; limitation, remedial, or compensatory.

The addition of the description of the type of action that may be taken is acceptable because it is describing the intent of the CTS SFDP purpose. This change is designated as administrative because it does not result in technical changes to the CTS.

A09 Unit 1 CTS 6.8.4.s and Unit 2 CTS 6.8.4.t format regarding the SFDP refers to remedial actions with their associated prescribed period for completion as "ACTION" and "allowed outage time." ITS 5.5.12 format refers to these same remedial action and prescribed completion periods as "Condition and Required Action" and "Completion Time." This changes the CTS by changing how the remedial actions and associated period for action completion is referred to consistent with the ITS format. |R1

The change in format of these remedial action requirements is acceptable because it continues to portray the intent of the CTS. This change is designated as administrative because it does not result in technical changes to the CTS.

A10 **Unit 1 only:** CTS 4.9.12.b.4 requires verifying a system flow rate of 10,350 cfm ± 10% for the required fuel pool area ventilation train during system operation when tested in accordance with ANSI N510-1975. ITS 5.5.8 does not explicitly require this verification because it is duplicative of the flow rate requirement specified in CTS 4.9.12.b.1 and b.2. This changes the CTS by deleting a duplicative requirement to verify fuel pool area ventilation train flowrate while operating the ventilation system. This change is designated as administrative because it does not result in technical changes to the CTS.

A11 The explosive gas mixture requirements in CTS 3.11.2.5 and the gas storage tank requirements of CTS 3.11.2.6 have been placed in a program in the proposed Administrative Controls Chapter 5.0 (ITS 5.5.9). As such, a general program statement has been added. Also, a statement of applicability of ITS SR 3.0.2 and SR 3.0.3 is needed to clarify the allowances for Surveillance Frequency extensions do apply. This changes the CTS by moving the explosive gas mixture and gas storage tank requirements to a program in ITS Section 5.5 and specifically stating the applicability of ITS SR 3.0.2 and SR 3.0.3 in the program.

The addition of the program statement is acceptable because it is describing the intent of the CTS Specification. The addition of ITS SR 3.0.2 and SR 3.0.3 statement is a clarification needed to maintain provisions that are currently allowed in the LCO and SR sections of the CTS, therefore it is considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

A12 **Unit 2 only:** CTS 4.6.6.1.b requires SBVS filter testing, in part, after any structural maintenance on the HEPA filter or charcoal adsorber housings; or following painting, fire, or chemical release in any ventilation zone communicating with the system. ITS 5.5.8 does not include these specific testing frequency requirements for the SBVS filters. CTS 6.8.4.k currently requires filter testing in accordance with Regulatory Guide (RG) 1.52, Revision 3. RG 1.52 contains the similar testing frequency requirements as stated in CTS 4.6.6.1.b. ITS 5.5.8 retains the requirement that the SBVS filter testing be performed, as applicable, in accordance with Regulatory Guide (RG) 1.52, Revision 3. Therefore, CTS 4.6.6.1.b testing frequency requirements specified herein are duplicative to those required by CTS 6.8.4.k (ITS 5.5.8) and are unnecessary to include in the ITS. As a result, this change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M01 **Unit 1 only:** CTS does not provide administrative controls for reactor RCP flywheel inspections. ITS 5.5.5, "Reactor Coolant Pump (RCP) Flywheel Inspection Program" is added to the ITS. The proposed program will provide for the inspection of each RCP flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975 except for an RCP flywheel consisting of ASTM A-516-69 (i.e., SA-516) Grade 65 material. For an RCP flywheel composed of ASTM A-516-69 Grade 65 material, the inspection shall consist of either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years. This changes the CTS by adding an additional administrative control program.

The purpose of the RCP flywheel program is to implement requirements to minimize the potential for failures of the flywheels of RCP motors in light-water-cooled power reactors. At PSL, RCP flywheels are interchanged between Unit 1 and Unit 2 during refurbishment and replacement. Between 2011 and 2017, the Unit 1 RCP flywheels, which consist of ASTM A-516-69, Grade 65 material, were exchanged with the Unit 2 RCP flywheels, which consist of ASTM A543 (three flywheels Class 1, Type B and one flywheel Grade 70) material. RCP flywheels with ASTM A543 material are inspected per Regulatory Guide 1.14, Revision 1, August 1975. As such, the added program will require the Unit 1 RCP flywheels to be inspected per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14 consistent with the ISTS. Additionally, the requirements specified for the PSL RCP flywheels composed of ASTM A-516-69 Grade 65 material are included in the ITS to support future RCP flywheel exchanges between Unit 1 and Unit 2. These requirements are consistent with the current requirements in Unit 2 CTS 6.8.4.o that were approved in Unit 2 Amendment No. 205, dated November 18, 2020 (ADAMS Accession No. ML20259A298). This change is designated as more restrictive because it imposes additional programmatic requirements in the Technical Specifications.

M02 CTS does not include a requirement for the Battery Monitoring and Maintenance Program. The ITS includes a requirement for this program. This changes the CTS by adding the ITS 5.5.14, "Battery Monitoring and Maintenance Program."

The Battery Monitoring and Maintenance Program is included to provide for battery restoration and maintenance per the guidance of IEEE-450. The specific wording associated with this program may be found in ITS 5.5.14. The Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (*76FR54510*) references the model application and safety evaluation for plant-specific adoption of TSTF–500, Revision 2 (NRC ADAMS Accession No. ML111751792). PSL has verified the applicable information specified in Section 2.2 of the TSTF-500 model application, including applicable UFSAR information. PSL will update the UFSAR, as necessary, to include any UFSAR information listed in Section 2.2 of the TSTF-500 model application that is not currently reflected in the PSL Unit 1 and Unit 2 UFSARs. This change is acceptable because it supports implementation of the requirements of the ITS. This change is designated as more restrictive because it imposes additional programmatic requirements in the Technical Specifications.

M03 CTS 6.14, in part, requires changes to the Offsite Dose Calculation Manual (ODCM) to be documented with a list of items contained in the documentation. ITS 5.5.1 also requires the same documentation and includes a requirement that records of reviews performed shall be retained. This change is necessary to incorporate the guidance related to ODCM changes specified in NRC Generic Letter GL 89-01, which states, "…the staff has concluded that records of licensee reviews performed for changes made to the ODCM… should be documented and retained for the duration of the unit operating license. This approach is in lieu of the current requirements that the reasons for changes to the ODCM… be addressed in the Semiannual Effluent Release Report." This change is designated as more restrictive because it imposes additional programmatic requirements in the Technical Specifications.

M04 Unit 1 CTS 6.8.4.h states, in part, that the peak calculated containment internal pressure for the design basis loss of coolant accident Pa, is 42.8 psig and Unit 2 CTS 6.8.4.h states, in part, that the peak calculated containment internal pressure for the design basis loss of coolant accident Pa, is 43.48 psig. Unit 1 ITS 5.5.13.b states, in part, that the calculated peak containment internal pressure for the design basis loss of coolant accident, Pa is 42.77 psig and Unit 2 ITS 5.5.13.b states, in part, that the calculated peak containment internal pressure for the design basis loss of coolant accident, Pa is 43.43 psig. This changes the CTS by revising the value of Pa to be consistent with the calculated peak containment internal pressure in the Unit 1 and Unit 2 analyses of record. R1

The purpose of the CTS requirement is to provide the plant specific value of Pa as defined in 10 CFR 50, Appendix J. The change is designated as more restrictive because it corrects a non-conservative Technical Specification value with respect to determining the allowable containment leakage limit, La.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 5 – Removal of SR Frequency to the Surveillance Frequency Control Program)* CTS 6.8.4.a contains a program requiring leakage testing of primary coolant sources outside containment with a stated frequency of every 18 months. CTS 6.8.4.k.5 contains a program requiring testing of the Shield Building Ventilation System (SBVS) filter heaters testing every 18 months. CTS 6.8.4.m.d requires measurement, at designated locations, of the control room envelope (CRE) pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation of the CREVS, operating at the flow rate required by the VFTP, at a frequency of 36 months on a Staggered Test Basis. CTS 6.8.4.n (iii) requires that total particulate concentration of the diesel generator fuel oil is ≤ 10 mg/l when tested every 31 days. ITS 5.5.2, 5.5.8, 5.5.15, and 5.5.10 contain similar requirements, respectively, but specify the periodic Frequency as "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified periodic Frequency for these tests to the Surveillance Frequency Control Program.

The purpose of these Surveillances is to assure that the necessary quality of systems and components is maintained. The removal of these details related to Surveillance Requirement Frequencies from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing Frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in NEI 04-10. The surveillance test requirements remain in the Technical Specifications. The control of changes to the Surveillance Frequencies is in accordance with the Surveillance Frequency Control Program, which is retained in ITS Section 5.5. The Surveillance Frequency Control Program provides the necessary administrative controls to require that surveillances related to testing, calibration and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met pursuant to the requirements of 10 CFR 50.36(c)(3). The proposed change to relocate periodic frequencies in the administrative controls section of Technical Specifications has been previously approved for Wolf Creek Generating Station Unit 1 in Amendment 227, dated April 8, 2021 (NRC ADAMS Accession No. ML21053A117), River Bend Station Unit 1 in Amendment 196, dated April 29, 2019 (NRC ADAMS Accession No. ML19066A008), and Grand Gulf Nuclear Station Unit 1 in Amendment 219, dated June 11, 2019 (NRC ADAMS Accession No. ML19094A799). PSL Unit 1 and Unit 2 adopted a Surveillance Frequency Control Program in Amendment Nos. 223 (Unit 1) and 173 (Unit 2) (ADAMS Accession No. ML15127A066) as contained in CTS 6.8.4.o (Unit 1) and 6.8.4.q (Unit 2) (ITS 5.5.16). This change is acceptable because the testing frequencies will be adequately controlled in accordance with the

Surveillance Frequency Control Program requirements retained in ITS, which ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change, because the Surveillance Frequencies are being removed from the Technical Specifications.

LA02 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 6.8.4.b, "In-Plant Radioiodine Monitoring," describes a program to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. ITS 5.5 does not include this program. This changes the CTS by moving the requirements for the In-Plant Radioiodine Monitoring Program to the Technical Requirements Manual (TRM).

The removal of this requirement from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The CTS 6.8.4.b program is designed to minimize radiation exposure to plant personnel in vital areas of the plant after an accident and has no impact on nuclear safety or the health and safety of the public. This change is acceptable because the program requirements will be adequately controlled in the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59, which ensure changes are properly evaluated. This change is designated as a less restrictive removal of requirement change because requirements are being removed from the Technical Specifications.

LA03 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.8.4 contains a Backup Method for Determining Subcooling Margin Program. This program ensures the capability to accurately monitor RCS subcooling margin by training and procedures. ITS 5.5 does not include this administrative program. This changes the CTS by moving administrative procedural detail to the updated Final Safety Analysis Report (UFSAR).

The removal of these administrative procedural details is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS retains the requirement to maintain the post accident monitoring instrument channels OPERABLE in MODES 1, 2, and 3 that are used to determine RCS subcooling margin and provide input into the RCS subcooling margin monitor (e.g. core exit thermocouples, RCS hot and cold leg temperature, and pressurizer pressure). Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA04 *(Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program)* CTS 6.8.4.g, "Radiological Environmental Monitoring Program," describes a program to monitor the radiation and radio-nuclides in the environs of the plant. ITS Section 5.5 does not require this program. This changes the CTS by moving the requirements for the Radiological Environmental Monitoring Program to the ODCM.

The purpose of these program requirements is to provide representative measurements of radioactivity in the highest potential exposure pathways, and verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The removal of the requirement for this program from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.6.2, “Radiological Effluent Release Report,” continues to require an annual report of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ODCM. Changes to the ODCM are controlled by the ODCM change control process specified in ITS 5.5.1, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of requirement change because the requirements for a program are being removed from the Technical Specifications.

LA05 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 4.8-2 footnote (c) states that the specified individual battery cell voltage limit for each connected cell is corrected for average electrolyte temperature. ITS 5.5.14 does not include the procedural detail regarding corrected cell voltage based on electrolyte temperature. This changes the CTS by moving procedural information from the Specification to the Battery Monitoring and Maintenance Program implementing document.

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.5.14 retains the requirement specifying the individual battery cell voltage limit. Also, this change is acceptable because these types of procedural details will be adequately controlled by the requirements of a program required by ITS Section 5.5. Changes to the program requirements will be controlled by the provisions of 10 CFR 50.59, which ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA06 *(Type 3 - Removing Procedural Detail for Meeting TS Requirements or Reporting Requirements)* CTS 3.11.2.5 includes the details for implementing the requirements for the explosive gas mixture. CTS 3.11.2.6 includes the details for implementing the requirements for the gas storage tanks. The details for implementing these requirements, including the specific limits for the explosive gas mixture and limit for quantity of radioactivity contained the gas storage tanks, are not included in the ITS. The ITS only includes a requirement to maintain a

program for these requirements. This changes the CTS by moving these procedural details for implementing the requirements, including the specific limits, from the Technical Specifications to the TRM.

The removal of these details for the specific explosive gas limits and limits on quantity of radioactivity in the gas storage tanks, including the Applicability, Actions, and Surveillance Requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 5.5.9 retains the requirement to include a program, which provides controls for potentially explosive gas mixtures contained in the waste gas decay tanks and the quantity of radioactivity contained in the gas storage tanks. Also, this change is acceptable because these types of procedural details will be adequately controlled in the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59, which ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA07 **Unit 2 only:** *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 4.6.6.1.b.1 requires a visual examination of the SBVS in accordance with ASME 510-1989. The ITS does not include this requirement. This changes the CTS by moving this requirement to the UFSAR.

The purpose of CTS 4.6.6.1.b is to ensure that in accordance with the Surveillance Frequency Control Program or; after any structural maintenance on the HEPA filter or charcoal adsorber housings; or following painting, fire, or chemical release in any ventilation zone communicating with the system, a visual inspection is performed on the SBVS. This is a good housekeeping practice that should be part of any recovery of a system from maintenance activities and periodically to visually identify gross system anomalies that would indicate the system is degraded. However, this detail is not necessary to be included in the ITS to provide adequate protection of the public health and safety. ITS 3.6.9 and ITS 5.5.8 retains Surveillance requirements and filter testing requirements to ensure the SBVS is OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR. Changes to the UFSAR are controlled by 10 CFR 50.59 or 10 CFR 50.71(e), which ensures that any changes to the UFSAR are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to meeting a TS requirement is being removed from the Technical Specifications.

LA08 (Type 4 – Removal of LCO, SR, or other TS Requirement to the TRM, UFSAR, ODCM, NQAP, CLRT Program, IST Program, or ISI Program) CTS 6.8.4.p, "Snubber Testing Program," describes a program that implements the 10 CFR 50.55a inservice inspection (ISI) requirements for dynamic restraints (snubbers). ITS Section 5.5 does not require this program. This changes the CTS by moving the requirements for the Snubber Testing Program to the TRM. R1

The removal of these requirements is acceptable because this type of information is not necessary to be included in the Technical Specifications to

provide adequate protection of public health and safety. The Technical Specifications retain the requirements for the affected components supported by the snubbers to be OPERABLE. ITS LCO 3.0.8 addresses the required actions when one or more required snubbers are found to be unable to perform their associated function. This change is acceptable because the program requirements will be adequately controlled in the TRM. Any changes to the TRM are made under 10 CFR 50.59, which ensures the changes are properly evaluated. This change is designated as a less restrictive removal of requirement change because requirements are being removed from the Technical Specifications. R1

LESS RESTRICTIVE CHANGES

L01 *(Category 7 – Relaxation of Surveillance Frequency)* CTS 6.8.4.l.d.2 requires, in part, inspecting 100% of the steam generator (SG) tubes at sequential periods of 144, 108, 72, and thereafter 60 effective full power months (EFPMs) and, in addition, requires inspecting 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected. ITS 5.5.6.d.2 requires inspection of 100% of the tubes in each steam generator at least every 96 EFPMs. CTS 6.8.4.l.d.3 requires additional inspection for each SG when crack indications are found in any tube at a frequency not to exceed 24 EFPMs or one refueling outage (whichever is less). ITS 5.5.6.d.3 requires the additional inspection on each affected and potentially affected SG at the next refueling outage. This changes the CTS by modifying the inspection frequency to a single requirement to inspect 100% of the SG tubes at a maximum frequency of 96 EFPMs and modifying the inspection frequency when crack indications are discovered to the next refueling outage.

The purpose of the inspection frequencies associated with the SG tubes is to ensure appropriate inspections are performed consistent with accepted NRC and industry practice as identified in Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines,” and its referenced Electric Power Research Institute SG examination guidelines, which establish the content of the Steam Generator Program. These guidelines minimize the potential of SG tube failures to support maintaining SG and reactor coolant pressure boundary structural integrity.

The proposed changes are consistent with the ISTS and Technical Specification Task Force (TSTF) traveler TSTF-577-A, Revision 1, “Revised Frequencies for Steam Generator Tube Inspections.” PSL Units 1 and 2 SGs contain Alloy 690 thermal treated (TT) tubes. For Alloy 690TT tubing, TSTF-577, which is incorporated in Revision 5 of the ISTS, revised the frequencies related to 100% and 50% inspection of the tubes such that both the maximum time between inspections and the time to inspect 100 percent of the tubes be 96 EFPMs. TSTF-577 also revised the frequency when crack indications are found in any tube to eliminate 24 EFPMs and require the inspection at the next refueling outage. The nuclear industry’s Steam Generator Task Force presented a technical basis supporting the 96 EFPM Alloy 690TT inspection interval during the February 13, 2019 (ADAMS Package Accession No. ML19044A416), and

February 24, 2020 (ADAMS Package Accession No. ML20066E421), public meetings with the NRC staff.

As stated in the NRC Safety Evaluation (SE) accompanying TSTF-577-A, proposed changes in TSTF-577 are acceptable because they continue to ensure SG tube integrity and, therefore, protect the public health and safety. In particular, the structural integrity performance criterion and accident-induced leakage performance criterion will continue to be met with the proposed revised SG inspection intervals (maximum allowable time between SG inspections) and inspection periods (maximum allowable time between 100 percent of SG tubes inspections). That considered, the changes to the SG tube inspection frequencies are acceptable. For Unit 1, inspection of 100% of the tubes in each SG was completed in the Spring 2021 refueling outage. For Unit 2, inspection of 100% of the tubes in each SG was completed for the current inspection period prior to the Fall 2021 refueling outage. Therefore, the 96 EFPM inspection period will begin from these SG tube inspection outages.

TSTF traveler TSTF-577-A incorporated changes to the Standard Technical Specifications (STSs) under the consolidated line item improvement process (CLIIP). TSTF-577-A was approved for use by the NRC as documented in the accompanying SE dated April 14, 2021 (ADAMS Accession No. ML21098A188, ML21096A274). PSL has reviewed the NRC SE and concluded that the justification presented in TSTF-577-A and the SE prepared by the NRC staff are applicable to PSL and justify this change.

This change is designated as less restrictive because the maximum inspection frequencies for the 100% and 50% tube inspections was extended to beyond the current inspection periods.

L02 **Unit 2 only:** (Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria) CTS 6.8.4.o requires administrative controls for reactor coolant pump (RCP) flywheel inspections. ITS 5.5.5 includes the same requirements and includes the CTS 6.8.4.o program requirements that were issued prior to Amendment No. 205 with a distinction on applicability of the two sets of requirements. The proposed program will provide for the inspection of each RCP flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975 except for an RCP flywheel consisting of ASTM A-516-69 (i.e., SA-516) Grade 65 material. For an RCP flywheel composed of ASTM A-516-69 Grade 65 material, the inspection shall consist of either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years. This changes the CTS by restoring program requirements that were issued prior to Amendment No. 205 in addition to the current requirements. |R1

The purpose of the RCP flywheel inspection program is to implement requirements to minimize the potential for failures of the flywheels of RCP motors in light-water-cooled power reactors. At PSL, RCP flywheels are interchanged between Unit 1 and Unit 2 during refurbishment and replacement. Between 2011

and 2017, the Unit 2 RCP flywheels, which consist of ASTM A-543 (three flywheels Class 1, Type B and one flywheel Grade 70) material, were exchanged with the Unit 1 RCP flywheels, which consist of ASTM A-516-69, Grade 65 material. Unit 2 CTS 6.8.4.o was modified to relax the RCP flywheel inspection requirements similar to the conclusions specified in the NRC Safety Evaluation of Topical Report SIR-94-080, Revision 1. The relaxed inspection requirements were approved in Amendment No. 205, dated November 18, 2020 (ADAMS Accession No. ML20259A298). Previously, PSL Unit 2 RCP flywheels were inspected per NRC Regulatory Guide 1.14, “Reactor Coolant Pump Flywheel Integrity,” Revision 1 and the requirements were previously contained in the CTS pre-Amendment 205 requirements. NRC Regulatory Guide 1.14 describes a method acceptable to the NRC staff of implementing RCP flywheel inspection requirements and include a recommended inspection schedule. Since the PSL RCP flywheels consisting of ASTM A-543 material were previously inspected per Regulatory Guide 1.14, Revision 1, August 1975 and the requirements were previously approved for use for Unit 2, the pre-Amendment 205 requirements are proposed to be restored in support of future RCP flywheel exchanges between Unit 1 and Unit 2. This change is designated as less restrictive because RCP flywheel inspection requirements, which have been previously approved for use at PSL, have been added.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

Definition 1.18

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification [5.6.1] and Specification [5.6.2]. (2)

Licensee initiated changes to the ODCM:

6.14.1 DOC M03

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations,

6.14.2 b. Shall become effective after the approval of the plant manager, and

6.14.3 c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

6.8.4.a

5.5.2 Primary Coolant Sources Outside Containment

High Pressure Safety Injection, Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [Low Pressure Injection, ~~Reactor Building~~ Spray~~, Makeup and Purification~~, and ~~Hydrogen Recombiner~~]. The program shall include the following:

Safety

Reactor Coolant Sampling

2

a. Preventive maintenance and periodic visual inspection requirements and

b. Integrated leak test requirements for each system at ~~least once per [18] months~~.

a Frequency in accordance with the Surveillance Frequency Control Program

3

The provisions of SR 3.0.2 are applicable.

~~[5.5.3 Post Accident Sampling~~

2

~~REVIEWER'S NOTE~~

~~This program may be eliminated based on the implementation of Topical Report CE NPSD-1157, Rev. 1, "Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Basis for CEOG Utilities," and the associated NRC Safety Evaluation dated May 16, 2000, and implementation of the following commitments:~~

~~1. [Licensee] has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. The contingency plans will be contained in emergency plan implementing procedures and implemented with the implementation of the License amendment. Establishment of contingency plans is considered a regulatory commitment.~~

~~2. The capability for classifying fuel damage events at the Alert level threshold has been established for [PLANT] at radioactivity levels of 300 mCi/cc dose equivalent iodine. This capability may utilize the normal sampling system and/or correlations of sampling or letdown line dose rates to coolant concentrations. This capability will be described in emergency plan implementing procedures and implemented with the implementation of the License amendment. The capability for classifying fuel damage events is considered a regulatory commitment.~~

~~3. [Licensee] has established the capability to monitor radioactive iodines that have been released to offsite environs. This capability is described in our emergency plan implementing procedures. The capability to monitor radioactive iodines is considered a regulatory commitment..~~

~~This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents~~

5.5 Programs and Manuals

~~5.5.3 Post Accident Sampling (continued)~~

~~and containment atmosphere samples under accident conditions. The program shall include the following:~~

~~a. Training of personnel,~~

~~b. Procedures for sampling and analysis, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.]~~

2
4

6.8.4.f

5.5.~~4~~ Radioactive Effluent Controls Program

3

4

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402,

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I,

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days,

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I,

5.5 Programs and Manuals

6.8.4.f 5.5.~~4~~ Radioactive Effluent Controls Program (continued) 4

3

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ,

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I,

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

6.8.4.q 5.5.~~5~~ Component Cyclic or Transient Limit

4

U 1 4 2 5.2

This program provides controls to track the FSAR, Section [~~ ~~], cyclic and transient occurrences to ensure that components are maintained within the design limits.

~~[5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program~~ 2 4

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.~~

~~The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]~~

5.5 Programs and Manuals

(RCP)

1

5.5.~~7~~ Reactor Coolant Pump Flywheel Inspection Program

4

DOC M01

5

RCP

1

This program shall provide for the inspection of each ~~reactor coolant pump~~ flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

Insert 1

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5.5.~~8~~ Steam Generator (SG) Program

4

6

6.8.4.l An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

6.8.4.l.a a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging ~~[or repair]~~ of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, ~~[~~or ~~repaired]~~ to confirm that the performance criteria are being met.

2

6.8.4.l.b b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

6.8.4.l.b.1 1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

6.8.4.l.b.2 2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for

CTS

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Insert 1

DOC M01 , except for an RCP flywheel consisting of ASTM A-516-69 (i.e., SA-516) Grade 65 material.

For an RCP flywheel composed of ASTM A-516-69 Grade 65 material, the inspection shall consist of either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years.

5.5 Programs and Manuals

5.5.~~8~~ 6 Steam Generator (SG) Program (continued) 4

6.8.4.l all SGs and leakage rate for an individual SG. Leakage is not to exceed ~~[1 gpm] per~~ SG. ~~[, except for specific types of degradation at specific locations as described in paragraph c of the SG Program.]~~

0.5 gpm total through all SGs and 0.25 gpm through any one

6.8.4.l.b.3 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

6.8.4.l.c c. Provisions for SG tube plugging ~~[or repair]~~ criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding ~~[~~40%~~]~~ of the nominal tube wall thickness shall be plugged ~~[or repaired]~~.

~~REVIEWER'S NOTE~~

~~Alternate tube plugging [or repair] criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube plugging [or repair] criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube plugging [or repair] criteria.~~ 6

~~[The following alternate tube plugging [or repair] criteria may be applied as an alternative to the 40% depth based criteria:~~ 2

~~1. . . .]~~

~~REVIEWER'S NOTE~~

~~The bracketed phrase in Paragraph d regarding exempt portions of the tube is only applicable to SGs with Alloy 600 thermally treated tubing.~~ 6

6.8.4.l.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet ~~[except for any portions of the tube that are exempt from inspection by alternate repair criteria]~~, and that may satisfy the applicable tube plugging ~~[or repair]~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

5.5 Programs and Manuals

5.5.~~8~~ 6 Steam Generator (SG) Program (continued) 4

~~REVIEWER'S NOTE~~

~~Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.~~ 6

6.8.4.l.d.1

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

~~[2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 24 effective full power months, which defines the inspection period.]~~

~~[2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 54 effective full power months, which defines the inspection period. If none of the SG tubes have ever experienced cracking other than in regions that are exempt from inspection by alternate repair criteria and the SG inspection was performed with enhanced probes, the inspection period may be extended to 72 effective full power months. Enhanced probes have a capability to detect flaws of any type equivalent to or better than array probe technology. The enhanced probes shall be used from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet except any portions of the tube that are exempt from inspection by alternate repair criteria. If there are regions where enhanced probes cannot be used, the tube inspection techniques shall be capable of detecting all forms of existing and potential degradation in that region.]~~

6.8.4.l.d.2
DOC L01

~~[~~2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period. ~~]~~ 2

~~REVIEWER'S NOTE~~

~~The bracketed phrases in Paragraph 3 are only applicable to SGs with Alloy 600 thermally treated tubing.~~ 6

6.8.4.l.d.3

DOC L01

3. If crack indications are found in any SG tube ~~[excluding any region that is exempt from inspection by alternate repair criteria]~~, then the next inspection for each ~~affected and potentially affected~~ SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage ~~[, but may be deferred to the following refueling~~ 2 1

5.5 Programs and Manuals

5.5.~~8~~ 6 Steam Generator (SG) Program (continued)

outage if the 100% inspection of all SGs was performed with enhanced probes as described in paragraph d.2]. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

6.8.4.l.d.3

6.8.4.l.e

e. Provisions for monitoring operational primary to secondary LEAKAGE.

[f. Provisions for SG tube repair methods. SG tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

---------- REVIEWER'S NOTE ----------

Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be used.

---

1. . . .]

6.8.4.c 5.5.~~9~~ 7 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include:

6.8.4.c (i) a. Identification of a sampling schedule for the critical variables and control points for these variables,

6.8.4.c (ii) b. Identification of the procedures used to measure the values of the critical variables,

6.8.4.c (iii) c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage,

6.8.4 (iv) d. Procedures for the recording and management of data,

6.8.4 (v) e. Procedures defining corrective actions for all off control point chemistry conditions, and

5.5 Programs and Manuals

5.5.~~9~~ 7 Secondary Water Chemistry Program (continued) 4

6.8.4 (vi)

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.~~10~~ 8 Ventilation Filter Testing Program (VFTP) 4

6.8.4 k

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems ~~at the frequencies specified in [Regulatory Guide], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1] at the system flowrate specified below [± 10%]~~. 2 1

Insert 2

6.8.4 k.1

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ≤ ~~[~~0.05~~]~~% when tested in accordance with ~~[Regulatory Guide 1.52, Revision 2, and~~ ASME N510-1989~~]~~ at the system flowrate specified below ~~[~~± 10%~~]~~. 1

ESF Ventilation System	Flowrate
~~[ ]~~	~~[ ]~~ Insert 3 ≤

6.8.4 k.2

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ ~~[~~0.05~~]~~% when tested in accordance with ~~[Regulatory Guide 1.52, Revision 2, and~~ ASME N510-1989~~]~~ at the system flowrate specified below ~~[~~± 10%~~]~~. 1 2

Insert 4

6.8.4 k.3

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in ~~[~~Regulatory Guide 1.52, Revision ~~2]~~, shows the methyl iodide penetration ~~less than the value specified below~~ ≤ 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and ~~the~~ relative humidity ~~specified below~~. 3 1

a of 70%

ESF Ventilation System	Penetration	RH	Face Velocity
CREVS SBVS ECCS Area Ventilation System	~~[ ]~~	~~[See Reviewer's Note]~~	~~[See Reviewer's Note]~~	~~[See Reviewer's Note]~~

~~REVIEWER'S NOTE~~

~~The use of any standard other than ASTM D3803-1989 to test the charcoal sample may result in an overestimation of the capability of the charcoal to adsorb radioiodine. As a result, the ability of the charcoal filters to perform in a manner consistent with the licensing basis for the facility is indeterminate.~~ 6

CTS

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Insert 2

6.8.4.k
4.9.12.b

and the Fuel Pool Area Ventilation System in accordance with Regulatory Guide 1.52, Revision 3, ASME N510-1989, ASTM D3803-1989 and, ANSI N510-1975, as described herein.

6.8.4.k

The tests described in Specification 5.5.8.a through 5.5.8.d shall be performed at the frequencies specified in Regulatory Guide 1.52, Revision 3.

4.9.12.b
4.9.12.c
4.6.6.1.c
4.7.8.1.c

The tests described in Specification 5.5.8.e through 5.5.8.k shall be performed at a frequency in accordance with the Surveillance Frequency Control Program.

4.9.12.b

The tests described in Specification 5.5.8.g through 5.5.8.i shall be performed after any structural maintenance on the HEPA filter or charcoal adsorber housings; and following painting, fire or chemical release in any ventilation zone communicating with the system.

4.9.12.d
4.9.12.e

The tests described in Specification 5.5.8.g and 5.5.8.h shall be performed after each complete or partial replacement of a HEPA filter bank.

Insert 3

6.8.4.k..1

Control Room Emergency Ventilation System (CREVS)	2000
Shield Building Ventilation System (SBVS)	6000
Emergency Core Cooling System (ECCS) Area Ventilation System	30,000

Insert 4

6.8.4.k.2

ESF Ventilation System	Flowrate
CREVS	2000
SBVS	6000
ECCS Area Ventilation System	30,000

5.5 Programs and Manuals

5.5.~~10~~ Ventilation Filter Testing Program (continued)

8

~~ASTM D 3803-1989 is a more stringent testing standard because it does not differentiate between used and new charcoal, it has a longer equilibration period performed at a temperature of 30°C (86°F) and a relative humidity (RH) of 95% (or 70% RH with humidity control), and it has more stringent tolerances that improve repeatability of the test.~~

~~Allowable Penetration = [(100% - Methyl Iodide Efficiency * for Charcoal Credited in Licensee's Accident Analysis) / Safety Factor]~~

~~When ASTM D3803-1989 is used with 30°C (86°F) and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:~~

~~Safety factor ≥ 2 for systems with or without humidity control.~~

~~Humidity control can be provided by heaters or an NRC-approved analysis that demonstrates that the air entering the charcoal will be maintained less than or equal to 70 percent RH under worst case design basis conditions.~~

~~If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified.~~

~~*This value should be the efficiency that was incorporated in the licensee's accident analysis which was reviewed and approved by the staff in a safety evaluation.~~

~~---~~

4

6

6.8.4 k.4

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters~~, the prefilters,~~ and the charcoal adsorbers is less than the value specified below when tested ~~in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989]~~ at the system flowrate specified below ~~[~~± 10%~~]~~.

ESF Ventilation System	Delta P	Flowrate
[ ]	[ ]	[ ]

Insert 5

1

f

6.8.4 k.5

~~[~~e. Demonstrate that the ~~heaters for each of the ESF systems dissipate the value specified below [± 10%]~~ when tested in accordance with ~~[~~ASME N510-1989~~]~~.

Insert 6

2

1

~~ESF Ventilation System Wattage]~~

~~[ ] [ ]~~

Insert 7

1

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

CTS

1

Insert 5

6.8.4.k.4

ESF Ventilation System	Delta P (in wg)	Flowrate
CREVS	< 4.15	2000
SBVS	≤ 6.15	6000
ECCS Area Ventilation System	< 4.15	30,000

4.6.6.1.c.1
4.7.8.1.c.1

e. Demonstrate for the SBVS and ECCS Area Ventilation System that the air flow distribution across HEPA filters and charcoal adsorbers is uniform within 20% when tested in accordance with ASME N510-1989.

Insert 6

6.8.4.k.5

SBVS main heaters dissipate 30 kW ± 10% and the SBVS auxiliary heaters dissipate ≥ 1.25 kW and ≤ 1.75 kW

CTS

1

Insert 7

4.9.12.b.2
4.9.12.d

g. Demonstrate for the Fuel Pool Area Ventilation System that an inplace test of the HEPA filters shows a removal efficiency of ≥ 99% of dioctyl phthalate when tested in accordance with ANSI N510-1975 at a system flowrate of 10,350 cfm ± 10%.

4.9.12.b.1
4.9.12.e

h. Demonstrate for the Fuel Pool Area Ventilation System that an inplace test of the charcoal adsorbers shows a removal efficiency ≥ 99% of a halogenated hydrocarbon refrigerant test gas when tested in accordance with ANSI N510-1975 at a system flowrate of 10,350 cfm ± 10%.

4.9.12.b.3

i. Demonstrate for the Fuel Pool Area Ventilation System that a laboratory test of a sample of the charcoal adsorber, when obtained as described herein, shows a removal efficiency of ≥ 85% for radioactive methyl iodide when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and relative humidity of 95%.

A carbon sample shall be obtained from either one test canister or two carbon samples removed from one of the charcoal adsorbers. Carbon samples not obtained from test canisters shall be prepared by emptying:

4.9.12.b.3 a)

1. one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

4.9.12.b.3 b)

2. a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4.9.12.c.1

j. Demonstrate for the Fuel Pool Area Ventilation System that the pressure drop across the combined HEPA filters and the charcoal adsorbers is < 4.15 inches water gauge when tested at a system flowrate of 10,350 cfm ± 10%.

4.9.12.c.2

k. Demonstrate for the Fuel Pool Area Ventilation System that the air flow distribution across HEPA filters and charcoal adsorbers is uniform within 20% when tested in accordance with ANSI N510-1975.

5.5 Programs and Manuals

5.5.~~11~~ Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the ~~[Waste Gas Holdup System], [~~the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]~~. ~~The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].~~
The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the ~~[Waste Gas Holdup System]~~ and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion),

b. A surveillance program to ensure that the quantity of radioactivity contained in ~~[~~each gas storage tank ~~and fed into the offgas treatment system]~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of ~~[~~an uncontrolled release of the tanks' contents~~]~~, and

~~c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5 Programs and Manuals

5.5.~~12~~ Diesel Fuel Oil Testing Program

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6.8.4.n

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

6.8.4.n (i)

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. A clear and bright appearance with proper color or a water and sediment content within limits,

6.8.4.n (ii)

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

6.8.4.n (iii)

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested ~~every 31 days~~ at a Frequency in accordance with the Surveillance Frequency Control Program.

3

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

6.8.4.j

5.5.~~13~~ Technical Specifications (TS) Bases Control Program

4

11

This program provides a means for processing changes to the Bases of these Technical Specifications.

6.8.4.j.1

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

6.8.4.j.2

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

6.8.4.j.2.a

1. A change in the TS incorporated in the license or

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6.8.4.j.2.b

2. A change to the ~~updated~~ FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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6.8.4.j.3

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

5.5 Programs and Manuals

6.8.4.j 5.5.~~13~~ 11 Technical Specifications (TS) Bases Control Program (continued) 4

6.8.4.j.4 d. Proposed changes that meet the criteria of 5.5.~~13~~11.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.~~14~~ 12 Safety Function Determination Program (SFDP) 4

6.8.4.s This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists,

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities, and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

5.5 Programs and Manuals

6.8.4.s 5.5.~~14~~ 12 Safety Function Determination Program (SFDP) (continued) 4

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

6.8.4.h 5.5.~~15~~ 13 Containment Leakage Rate Testing Program 4

~~[OPTION A]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~

~~b. The maximum allowable containment leakage rate, L_a at P_a, shall be [ ]% of containment air weight per day.~~

~~c. Leakage rate acceptance criteria are:~~

~~1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and < 0.75 L_a for Type A tests.~~ 8

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.~~

~~b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥ 10 psig].~~

~~d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~

5.5 Programs and Manuals

6.8.4.h 5.5.~~15~~ Containment Leakage Rate Testing Program (continued)

13

~~[OPTION B]~~

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6.8.4.h a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:~~

NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.~~

~~[3. . . .]~~

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6.8.4.h b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is ~~[45~~ psig~~]~~. The containment design pressure is ~~[50~~ psig~~]~~.

42.77

44

2 | R1

6.8.4.h c. The maximum allowable containment leakage rate, L_a at P_a, shall be [ ]% of containment air weight per day.

0.50

2

6.8.4.h d. Leakage rate acceptance criteria are:

6.8.4.h.a 1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

6.8.4.h.b 2. Air lock testing acceptance criteria are:

personnel air lock

a) Overall air lock leakage rate is ≤ ~~[~~0.05 L_a~~]~~ when tested at ≥ P_a.

b) For each door, leakage rate is ~~≤~~< ~~[~~0.01 L_a~~]~~ when pressurized to ~~[~~≥ ~~10 psig]~~.

1.0 P_a

c) For each emergency air lock door, leakage rate is < 0.01 L_a when pressurized to ≥ 10 psig.

1

2

1

5.5 Programs and Manuals

5.5.~~15~~ 13 Containment Leakage Rate Testing Program (continued) 4

6.8.4.h e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.8.4.h f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

~~[OPTION A/B Combined]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C] [Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:~~

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~

8

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.~~

~~[3. . . .]~~

~~b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is [45 psig]. The containment design pressure is [50 psig].~~

~~c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.~~

5.5 Programs and Manuals

~~5.5.15 Containment Leakage Rate Testing Program (continued)~~

~~d. Leakage rate acceptance criteria are:~~

~~1. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $<$ 0.60 L_a for the Type B and C tests and [$<$ 0.75 L_a for Option A Type A tests] [$\leq$ 0.75 L_a for Option B Type A tests].~~

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is $\leq$ [0.05 L_a] when tested at $\geq P_a$.~~

~~b) For each door, leakage rate is $\leq$ [0.01 L_a] when pressurized to [$\geq$ 10 psig].~~

~~e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~

8

5.5.~~16~~ Battery Monitoring and Maintenance Program

14

4

~~REVIEWER'S NOTE~~

~~This program and the corresponding requirements in LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6 require providing the information and verifications requested in the Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (76FR54510).~~

6

DOC M02

This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-~~2002~~2010, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed by Regulatory Guide 1.129, Revision ~~2 (RG), with RG exceptions and program provisions as identified below:~~ 3.

~~a. The program allows the following RG 1.129, Revision 2 exceptions:~~

~~1. Battery temperature correction may be performed before or after conducting discharge tests.~~

~~2. RG 1.129, Regulatory Position 1, Subsection 2, "References," is not applicable to this program.~~

9

5.5 Programs and Manuals

5.5.~~16~~ Battery Monitoring and Maintenance Program (continued) 4

14

~~3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, "Inspections," the following shall be used: "Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery."~~

~~4 In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."~~

~~5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."~~ 9

~~b.~~ The program shall include the following provisions:

Table 4.8-2

a. ~~1.~~ Actions to restore battery cells with float voltage < ~~[~~2.13~~]~~ V;

Table 4.8-2

b. ~~2.~~ Actions to determine whether the float voltage of the remaining battery cells is ≥ ~~[~~2.13~~]~~ V when the float voltage of a battery cell has been found to be < ~~[~~2.13~~]~~ V; 2

c. ~~3.~~ Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

d. ~~4.~~ Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

~~5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.~~ 9

5.5 Programs and Manuals

6.8.4.m 5.5.~~17~~ 15 Control Room Envelope (CRE) Habitability Program 4

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency ~~Air Cleanup~~ System (CRE~~AC~~S), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of ~~[5 rem whole body or its equivalent to any part of the body] [~~5 rem total effective dose equivalent (TEDE)~~]~~ for the duration of the accident. The program shall include the following elements:

Ventilation V 1 2

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

~~[The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:~~

~~1. ;and]~~ 2

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the ~~CRE~~ACS, operating at the flow rate required by the VFTP, at a Frequency ~~of [18] months on a STAGGERED TEST BASIS~~. The results shall be trended and used as part of the ~~[18]~~ month assessment of the CRE boundary.

CREVS | R1 3

in accordance with the Surveillance Frequency Control Program. 36 2

5.5 Programs and Manuals

6.8.4.m 5.5.~~17~~ Control Room Envelope (CRE) Habitability Program (continued) 4

15

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

~~[5.5.18 Setpoint Control Program~~

~~REVIEWER'S NOTE~~

~~Adoption of a Setpoint Control Program requires changes to other technical specifications. See TSTF-493, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions," Option B, for guidance (Agencywide Documents Access and Management System (ADAMS) Accession Number ML101160026).~~

~~This program shall establish the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analyses, provides a means for processing changes to instrumentation setpoints, and identifies setpoint methodologies to ensure instrumentation will function as required. The program shall ensure that testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verifies that instrumentation will function as required.~~

10

~~a. The program shall list the Functions in the following specifications to which it applies:~~

~~1. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating [(Analog)] [(Digital)];"~~

5.5 Programs and Manuals

5.5.18 Setpoint Control Program (continued)

2. LCO 3.3.2, "Reactor Protective System (RPS) Instrumentation - Shutdown [(Analog)] [(Digital)];"
3. LCO [3.3.3, "Control Element Assembly Calculators (CEACs) (Digital)];"
4. [LCO 3.3.4,"Engineered Safety Features Actuation System (ESFAS) Instrumentation (Analog);"] [LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Instrumentation (Digital);"]
5. [LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS) (Analog);"] [LCO 3.3.7, "Diesel Generator (DG) - Loss of Voltage Start (LOVS) (Digital);"]
6. [LCO 3.3.7, "Containment Purge Isolation Signal (CPIS) (Analog);"] [LCO 3.3.8, "Containment Purge Isolation Signal (CPIS) (Digital);"]
7. [LCO 3.3.8, "Control Room Isolation Signal (CRIS) (Analog);"] [LCO 3.3.9, "Control Room Isolation Signal (CRIS) (Digital);"];
8. [LCO 3.3.9, "Chemical and Volume Control System (CVCS) Isolation Signal (Analog);"]
9. [LCO 3.3.10, "Fuel Handling Isolation Signal (FHIS) (Digital);"]
10. LCO 3.3.13, "[Logarithmic] Power Monitoring Channels [(Analog)."] [(Digital)."]

b. The program shall require the [Limiting Trip Setpoint (LTSP)], [Nominal Trip Setpoint (NTSP)], Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) of the Functions described in paragraph a. are calculated using the NRC approved setpoint methodology, as listed below. In addition, the program shall contain the value of the [LTSP], [NTSP], AV, AFT, and ALT (as applicable) for each Function described in paragraph a. and shall identify the setpoint methodology used to calculate these values.

---------- Reviewer's Note ----------
List the NRC safety evaluation report by letter, date, and ADAMS accession number (if available) that approved the setpoint methodologies.
---

1. [Insert reference to NRC safety evaluation that approved the setpoint methodology.]

c. The program shall establish methods to ensure that Functions described in paragraph a. will function as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.

5.5 Programs and Manuals

5.5.18 Setpoint Control Program (continued)

d. ----------REVIEWER'S NOTE----------
A license amendment request to implement a Setpoint Control Program must list the instrument functions to which the program requirements of paragraph d. will be applied. Paragraph d. shall apply to all Functions in the Reactor Protection System and Engineered Safety Feature Actuation System specifications unless one or more of the following exclusions apply:

1. Manual actuation circuits, automatic actuation logic circuits or to instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.

2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.

3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.

The program shall identify the Functions described in paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The [LTSP] of these Functions are Limiting Safety System Settings. These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS and CHANNEL FUNCTIONAL TESTS that verify the [LTSP or NTSP].

1. The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified [LTSP or NTSP].

2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified [LTSP or NTSP] by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.

5.5 Programs and Manuals

~~5.5.18 Setpoint Control Program (continued)~~

~~3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.~~

~~4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the [LTSP or NTSP] at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the [LTSP or NTSP] provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).~~

10

~~e. The program shall be specified in [insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference].]~~

~~[~~ 5.5.~~19~~

16

6.8.4.o

Surveillance Frequency Control Program

2

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This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program. ~~]~~

2

~~[~~ 5.5.~~20~~

17

6.8.4.r

Risk Informed Completion Time Program

2

10

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

5.5 Programs and Manuals

6.8.4.r 5.5.~~20~~ Risk Informed Completion Time Program (continued)

17

10

~~REVIEWER'S NOTE~~

~~The Risk Informed Completion Time is only applicable in MODES supported by the licensee's PRA. Licensees applying the RICT Program to MODES other than MODES 1 and 2 must demonstrate that they have the capability to calculate a RICT in those MODES or that the risk indicated by their MODE 1 and 2 PRA model is bounding with respect to the lower MODE conditions.~~

6

MODES 1 and 2

b. A RICT may only be utilized in ~~MODE 1, 2 [, and 3, and MODE 4 while relying on steam generators for heat removal]~~;

2

c. When a RICT is being used, any change to the plant configuration~~, as defined in NEI 06-09-A, Appendix A,~~ must be considered for the effect on the RICT.

within the scope of the Risk Informed Completion Time Program

1

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

e

d. ~~For emergent conditions,~~ If the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

1

1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

d. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

1

5.5 Programs and Manuals

5.5.20 Risk Informed Completion Time Program (continued)

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.] 1

[5.5.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

This Program provides controls for monitoring the condition of the neutron absorber used in the spent fuel pool storage racks to verify the Boron-10 areal density is consistent with the assumptions in the spent fuel pool criticality analysis. The program shall be in accordance with NEI 16-03-A, "Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools," Revision 0, May 2017 [, with the following exceptions:

1. . . .].] 2

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

Definition 1.18

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification ~~[~~5.6.1~~]~~ and Specification ~~[~~5.6.2~~]~~. (2)

Licensee initiated changes to the ODCM:

6.14.1 DOC M03

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations,

6.14.2 b. Shall become effective after the approval of the plant manager, and

6.14.3 c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

6.8.4.a

5.5.2 Primary Coolant Sources Outside Containment

High Pressure Safety Injection, Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [Low Pressure Injection, ~~Reactor Building~~ Spray~~, Makeup and Purification~~, and ~~Hydrogen Recombiner~~]. The program shall include the following:

Safety

Reactor Coolant Sampling

2

a. Preventive maintenance and periodic visual inspection requirements and

b. Integrated leak test requirements for each system at ~~least once per [18] months~~.

a Frequency in accordance with the Surveillance Frequency Control Program

3

The provisions of SR 3.0.2 are applicable.

~~[5.5.3 Post Accident Sampling~~

2

~~REVIEWER'S NOTE~~

~~This program may be eliminated based on the implementation of Topical Report CE NPSD-1157, Rev. 1, "Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Basis for CEOG Utilities," and the associated NRC Safety Evaluation dated May 16, 2000, and implementation of the following commitments:~~

~~1. [Licensee] has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. The contingency plans will be contained in emergency plan implementing procedures and implemented with the implementation of the License amendment. Establishment of contingency plans is considered a regulatory commitment.~~

~~2. The capability for classifying fuel damage events at the Alert level threshold has been established for [PLANT] at radioactivity levels of 300 mCi/cc dose equivalent iodine. This capability may utilize the normal sampling system and/or correlations of sampling or letdown line dose rates to coolant concentrations. This capability will be described in emergency plan implementing procedures and implemented with the implementation of the License amendment. The capability for classifying fuel damage events is considered a regulatory commitment.~~

~~3. [Licensee] has established the capability to monitor radioactive iodines that have been released to offsite environs. This capability is described in our emergency plan implementing procedures. The capability to monitor radioactive iodines is considered a regulatory commitment..~~

~~This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents~~

5.5 Programs and Manuals

~~5.5.3 Post Accident Sampling (continued)~~

~~and containment atmosphere samples under accident conditions. The program shall include the following:~~

~~a. Training of personnel,~~

~~b. Procedures for sampling and analysis, and~~

~~c. Provisions for maintenance of sampling and analysis equipment.]~~

2
4

6.8.4.f

5.5.~~4~~ Radioactive Effluent Controls Program

3

4

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402,

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I,

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days,

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I,

5.5 Programs and Manuals

6.8.4.f 5.5.~~4~~ Radioactive Effluent Controls Program (continued) 4

3

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ,

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I,

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

6.8.4.r 5.5.~~5~~ Component Cyclic or Transient Limit 1 4 2

4

U

3.9

This program provides controls to track the FSAR, Section [~~ ~~], cyclic and transient occurrences to ensure that components are maintained within the design limits.

~~[5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program~~ 2 4

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.~~

~~The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.]~~

5.5 Programs and Manuals

5.5.~~7~~ Reactor Coolant Pump Flywheel Inspection Program

6.8.4.o
DOC L02

This program shall provide for the inspection of each ~~reactor coolant pump~~ flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

5.5.~~8~~ Steam Generator (SG) Program

6.8.4.l

An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

6.8.4.l.a

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging ~~[or repair]~~ of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, ~~[~~or ~~repaired]~~ to confirm that the performance criteria are being met.

6.8.4.l.b

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

6.8.4.l.b.1

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

6.8.4.l.b.2

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for

CTS

5

Insert 1

6.8.4.o
DOC L02

, except for an RCP flywheel consisting of ASTM A-516-69 (i.e., SA-516) Grade 65 material.

For an RCP flywheel composed of ASTM A-516-69 Grade 65 material, the inspection shall consist of either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years.

5.5 Programs and Manuals

5.5.~~8~~ 6 Steam Generator (SG) Program (continued) 4

6.8.4.l all SGs and leakage rate for an individual SG. Leakage is not to exceed ~~[1 gpm] per~~ SG. ~~[, except for specific types of degradation at specific locations as described in paragraph c of the SG Program.]~~

0.5 gpm total through all SGs and 0.25 gpm through any one

6.8.4.l.b.3 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

6.8.4.l.c c. Provisions for SG tube plugging ~~[or repair]~~ criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding ~~[~~40%~~]~~ of the nominal tube wall thickness shall be plugged ~~[or repaired]~~.

~~REVIEWER'S NOTE~~

~~Alternate tube plugging [or repair] criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube plugging [or repair] criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube plugging [or repair] criteria.~~ 6

~~[The following alternate tube plugging [or repair] criteria may be applied as an alternative to the 40% depth based criteria:~~ 2

~~1. . . .]~~

~~REVIEWER'S NOTE~~

~~The bracketed phrase in Paragraph d regarding exempt portions of the tube is only applicable to SGs with Alloy 600 thermally treated tubing.~~ 6

6.8.4.l.d d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet ~~[except for any portions of the tube that are exempt from inspection by alternate repair criteria]~~, and that may satisfy the applicable tube plugging ~~[or repair]~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

5.5 Programs and Manuals

5.5.~~8~~ 6 Steam Generator (SG) Program (continued) 4

6.8.4.l

~~REVIEWER'S NOTE~~

~~Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.~~ 6

6.8.4.l.d.1

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

~~[2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 24 effective full power months, which defines the inspection period.]~~

~~[2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 54 effective full power months, which defines the inspection period. If none of the SG tubes have ever experienced cracking other than in regions that are exempt from inspection by alternate repair criteria and the SG inspection was performed with enhanced probes, the inspection period may be extended to 72 effective full power months. Enhanced probes have a capability to detect flaws of any type equivalent to or better than array probe technology. The enhanced probes shall be used from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet except any portions of the tube that are exempt from inspection by alternate repair criteria. If there are regions where enhanced probes cannot be used, the tube inspection techniques shall be capable of detecting all forms of existing and potential degradation in that region.]~~

6.8.4.l.d.2
DOC L01

~~[~~2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period. ~~]~~ 2

~~REVIEWER'S NOTE~~

~~The bracketed phrases in Paragraph 3 are only applicable to SGs with Alloy 600 thermally treated tubing.~~ 6

6.8.4.l.d.3

DOC L01

3. If crack indications are found in any SG tube ~~[excluding any region that is exempt from inspection by alternate repair criteria]~~, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage ~~[, but may be deferred to the following refueling~~ 2 1

5.5 Programs and Manuals

5.5.~~8~~ 6 Steam Generator (SG) Program (continued)

~~outage if the 100% inspection of all SGs was performed with enhanced probes as described in paragraph d.2~~]. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

6.8.4.l.d.3

6.8.4.l.e

e. Provisions for monitoring operational primary to secondary LEAKAGE.

~~[f. Provisions for SG tube repair methods. SG tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.~~

~~REVIEWER'S NOTE~~

~~Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be used.~~

~~1. . . .]~~

6.8.4.c 5.5.~~9~~ 7 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation ~~and low pressure turbine disc stress corrosion cracking~~. The program shall include:

6.8.4.c (i)

a. Identification of a sampling schedule for the critical variables and control points for these variables,

6.8.4.c (ii)

b. Identification of the procedures used to measure the values of the critical variables,

6.8.4.c (iii)

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage,

6.8.4 (iv)

d. Procedures for the recording and management of data,

6.8.4 (v)

e. Procedures defining corrective actions for all off control point chemistry conditions, and

5.5 Programs and Manuals

5.5.~~9~~ 7 Secondary Water Chemistry Program (continued) 4

6.8.4 (vi) f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.~~10~~ 8 Ventilation Filter Testing Program (VFTP)

6.8.4 k A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems ~~at the frequencies specified in [Regulatory Guide], and~~ in accordance with ~~[~~Regulatory Guide 1.52, Revision ~~2, ASME N510-1989, and AG-1] at the system flowrate specified below [± 10%]~~. 4 2 1

Insert 2

6.8.4 k.1 a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ≤ ~~[~~0.05~~]~~% when tested in accordance with ~~[Regulatory Guide 1.52, Revision 2, and~~ ASME N510-1989~~]~~ at the system flowrate specified below ~~[~~± 10%~~]~~. 1

ESF Ventilation System	Flowrate
~~[ ]~~	~~[ ]~~ Insert 3 ≤

6.8.4 k.2 b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ ~~[~~0.05~~]~~% when tested in accordance with ~~[Regulatory Guide 1.52, Revision 2, and~~ ASME N510-1989~~]~~ at the system flowrate specified below ~~[~~± 10%~~]~~. 1 2

Insert 4

6.8.4 k.3 c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in ~~[~~Regulatory Guide 1.52, Revision ~~2]~~, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and ~~the~~ relative humidity ~~specified below~~. 3 1

a of 95%

ESF Ventilation System	~~Penetration~~	~~RH~~	~~Face Velocity~~
~~[ ]~~	~~[See Reviewer's Note]~~	~~[See Reviewer's Note]~~	~~[See Reviewer's Note]~~

Insert 5 1

~~REVIEWER'S NOTE~~

~~The use of any standard other than ASTM D3803-1989 to test the charcoal sample may result in an overestimation of the capability of the charcoal to adsorb radioiodine. As a result, the ability of the charcoal filters to perform in a manner consistent with the licensing basis for the facility is indeterminate.~~ 6

CTS

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Insert 2

6.8.4.k 3, ASME N510-1989, and ASTM D3803-1989, as described herein.

6.8.4.k The tests described in Specification 5.5.8.a through 5.5.8.d shall be performed at the frequencies specified in Regulatory Guide 1.52, Revision 3.

4.6.6.1.b The tests described in Specification 5.5.8.e shall be performed at the frequencies specified in Regulatory Guide 1.52, Revision 3, except that the testing frequency of 24 months is required at a frequency in accordance with the Surveillance Frequency Control Program.

6.8.4.k.5 The tests described in Specification 5.5.8.f shall be performed at a frequency in accordance with the Surveillance Frequency Control Program.

Insert 3

6.8.4.k..1

Control Room Emergency Ventilation System (CREVS)	2000
Shield Building Ventilation System (SBVS)	6000
Emergency Core Cooling System (ECCS) Area Ventilation System	30,000

Insert 4

6.8.4.k.2

ESF Ventilation System	Flowrate
CREVS	2000
SBVS	6000
ECCS Area Ventilation System	30,000

Insert 5

6.8.4.k.3

ESF Ventilation System	Penetration
CREVS	≤ 0.175%
SBVS	≤ 2.5%
ECCS Area Ventilation System	≤ 2.5%

5.5 Programs and Manuals

5.5.~~10~~ Ventilation Filter Testing Program (continued)

~~ASTM D 3803-1989 is a more stringent testing standard because it does not differentiate between used and new charcoal, it has a longer equilibration period performed at a temperature of 30°C (86°F) and a relative humidity (RH) of 95% (or 70% RH with humidity control), and it has more stringent tolerances that improve repeatability of the test.~~

~~Allowable Penetration = [(100% - Methyl Iodide Efficiency * for Charcoal Credited in Licensee's Accident Analysis) / Safety Factor]~~

~~When ASTM D3803-1989 is used with 30°C (86°F) and 95% RH (or 70% RH with humidity control) is used, the staff will accept the following:~~

~~Safety factor ≥ 2 for systems with or without humidity control.~~

~~Humidity control can be provided by heaters or an NRC-approved analysis that demonstrates that the air entering the charcoal will be maintained less than or equal to 70 percent RH under worst case design basis conditions.~~

~~If the system has a face velocity greater than 110 percent of 0.203 m/s (40 ft/min), the face velocity should be specified.~~

~~*This value should be the efficiency that was incorporated in the licensee's accident analysis which was reviewed and approved by the staff in a safety evaluation.~~

6.8.4 k.4

d. ~~Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].~~

Insert 6

~~ESF Ventilation System~~	~~Delta P~~	~~Flowrate~~
~~[ ]~~	~~[ ]~~	~~[ ]~~

6.8.4 k.5

~~[~~e. Demonstrate that the ~~heaters for each of the ESF systems dissipate the value specified below [± 10%]~~ when tested in accordance with ~~[~~ASME N510-1989~~]~~.

Insert 7

~~ESF Ventilation System~~	~~Wattage]~~
~~[ ]~~	~~[ ]~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

CTS

1

Insert 6

6.8.4.k.4

Demonstrate for each of the ESF systems, when tested at the system flowrate specified below ± 10%, that the pressure drop across the following system components is less than the value specified below:

1. For the CREVS and ECCS Area Ventilation System: combined HEPA filters and the charcoal adsorbers; and

2. For the SBVS, combined demisters, electric heaters, HEPA filters and the charcoal adsorbers.

ESF Ventilation System	Delta P (in wg)	Flowrate
CREVS	< 7.4	2000
SBVS	< 8.5	6000
ECCS Area Ventilation System	< 4.35	30,000

R1

4.6.6.1.b.2

e. Demonstrate for the SBVS that the air flow distribution across HEPA filters and charcoal adsorbers is uniform within 20% when tested in accordance with ASME N510-1989.

Insert 7

6.8.4.k.5

SBVS main heaters dissipate 30 kW ± 10% and the SBVS auxiliary heaters dissipate ≥ 1.25 kW and ≤ 1.75 kW

5.5 Programs and Manuals

5.5.~~11~~ Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the ~~[Waste Gas Holdup System], [~~the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]~~. ~~The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].~~
The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the ~~[Waste Gas Holdup System]~~ and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion),

b. A surveillance program to ensure that the quantity of radioactivity contained in ~~[~~each gas storage tank ~~and fed into the offgas treatment system]~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of ~~[~~an uncontrolled release of the tanks' contents~~]~~, and

~~c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste Treatment System] is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5 Programs and Manuals

5.5.~~12~~ Diesel Fuel Oil Testing Program

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6.8.4.n

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

6.8.4.n (i)

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. A clear and bright appearance with proper color or a water and sediment content within limits,

6.8.4.n (ii)

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

6.8.4.n (iii)

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested ~~every 31 days~~ at a Frequency in accordance with the Surveillance Frequency Control Program.

3

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

6.8.4.j

5.5.~~13~~ Technical Specifications (TS) Bases Control Program

11

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This program provides a means for processing changes to the Bases of these Technical Specifications.

6.8.4.j.1

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

6.8.4.j.2

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

6.8.4.j.2.a

1. A change in the TS incorporated in the license or

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U

6.8.4.j.2.b

2. A change to the ~~updated~~ FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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6.8.4.j.3

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

5.5 Programs and Manuals

6.8.4.j 5.5.~~13~~ 11 Technical Specifications (TS) Bases Control Program (continued) 4

6.8.4.j.4 d. Proposed changes that meet the criteria of 5.5.~~13~~11.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.~~14~~ 12 Safety Function Determination Program (SFDP) 4

6.8.4.t This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists,

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities, and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

5.5 Programs and Manuals

6.8.4.t 5.5.~~14~~ 12 Safety Function Determination Program (SFDP) (continued) 4

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

6.8.4.h 5.5.~~15~~ 13 Containment Leakage Rate Testing Program 4

~~[OPTION A]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~

~~b. The maximum allowable containment leakage rate, L_a at P_a, shall be [ ]% of containment air weight per day.~~

~~c. Leakage rate acceptance criteria are:~~

~~1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and < 0.75 L_a for Type A tests.~~ 8

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.~~

~~b) For each door, leakage rate is ≤ [0.01 L_a] when pressurized to [≥ 10 psig].~~

~~d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~

5.5 Programs and Manuals

6.8.4.h 5.5.~~15~~ Containment Leakage Rate Testing Program (continued)

13

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~~[OPTION B]~~

6.8.4.h a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:~~

NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 18, 2007 Type A test shall be performed no later than December 18, 2022.

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.~~

~~[3. . . .]~~

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6.8.4.h b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is ~~[45~~ psig~~]~~. The containment design pressure is ~~[50~~ psig~~]~~.

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43.43

2 | R1

6.8.4.h c. The maximum allowable containment leakage rate, L_a at P_a, shall be ~~[~~ ~~]~~% of containment air weight per day.

0.50

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6.8.4.h d. Leakage rate acceptance criteria are:

6.8.4.h.a 1. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

6.8.4.h.b 2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is ≤ ~~[~~0.05 L_a~~]~~ when tested at ≥ P_a.

b) For each door, leakage rate is ~~≤~~< ~~[~~0.01 L_a~~]~~ when pressurized to ~~[~~≥ ~~10 psig]~~.

1.0 P_a

2

5.5 Programs and Manuals

5.5.~~15~~ 13 Containment Leakage Rate Testing Program (continued) 4

6.8.4.h e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.8.4.h f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

~~[OPTION A/B Combined]~~

~~a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J. [Type A][Type B and C] test requirements are in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions. [Type B and C] [Type A] test requirements are in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The 10 CFR 50, Appendix J, Option B test requirements shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:~~

~~1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.~~

~~2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.~~

~~[3. . . .]~~

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~~b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a is [45 psig]. The containment design pressure is [50 psig].~~

~~c. The maximum allowable containment leakage rate, L_a, at P_a, shall be [ ]% of containment air weight per day.~~

5.5 Programs and Manuals

~~5.5.15 Containment Leakage Rate Testing Program (continued)~~

~~d. Leakage rate acceptance criteria are:~~

~~1. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $<$ 0.60 L_a for the Type B and C tests and [$<$ 0.75 L_a for Option A Type A tests] [$\leq$ 0.75 L_a for Option B Type A tests].~~

~~2. Air lock testing acceptance criteria are:~~

~~a) Overall air lock leakage rate is $\leq$ [0.05 L_a] when tested at $\geq P_a$.~~

~~b) For each door, leakage rate is $\leq$ [0.01 L_a] when pressurized to [$\geq$ 10 psig].~~

~~e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.~~

~~f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.~~

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5.5.~~16~~ Battery Monitoring and Maintenance Program

14

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~~REVIEWER'S NOTE~~

~~This program and the corresponding requirements in LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6 require providing the information and verifications requested in the Notice of Availability for TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360," (76FR54510).~~

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DOC M02

This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-~~2002~~2010, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed by Regulatory Guide 1.129, Revision ~~2 (RG), with RG exceptions and program provisions as identified below:~~ 3.

~~a. The program allows the following RG 1.129, Revision 2 exceptions:~~

~~1. Battery temperature correction may be performed before or after conducting discharge tests.~~

~~2. RG 1.129, Regulatory Position 1, Subsection 2, "References," is not applicable to this program.~~

9

5.5 Programs and Manuals

5.5.~~16~~ Battery Monitoring and Maintenance Program (continued)

14

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~~3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, "Inspections," the following shall be used: "Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery."~~

~~4 In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."~~

~~5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."~~

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~~b.~~ The program shall include the following provisions:

Table 4.8-2

a. ~~1.~~ Actions to restore battery cells with float voltage < ~~[~~2.13~~]~~ V;

Table 4.8-2

b. ~~2.~~ Actions to determine whether the float voltage of the remaining battery cells is ≥ ~~[~~2.13~~]~~ V when the float voltage of a battery cell has been found to be < ~~[~~2.13~~]~~ V;

2

c. ~~3.~~ Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

d. ~~4.~~ Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

~~5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.~~

9

5.5 Programs and Manuals

6.8.4.m 5.5.~~17~~ 15 Control Room Envelope (CRE) Habitability Program 4

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency ~~Air Cleanup~~ System (CRE~~AC~~S), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of ~~[5 rem whole body or its equivalent to any part of the body] [~~5 rem total effective dose equivalent (TEDE)~~]~~ for the duration of the accident. The program shall include the following elements:

Ventilation V 1 2

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

~~[The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:~~

~~1. ;and]~~ 2

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the ~~CREACS~~, operating at the flow rate required by the VFTP, at a Frequency ~~of [18] months on a STAGGERED TEST BASIS~~. The results shall be trended and used as part of the ~~[18]~~ month assessment of the CRE boundary. 36

in accordance with the Surveillance Frequency Control Program.

CREVS | R1 3 2

5.5 Programs and Manuals

6.8.4.m 5.5.~~17~~ Control Room Envelope (CRE) Habitability Program (continued) 4

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e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

~~[5.5.18 Setpoint Control Program~~

~~REVIEWER'S NOTE~~

~~Adoption of a Setpoint Control Program requires changes to other technical specifications. See TSTF-493, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions," Option B, for guidance (Agencywide Documents Access and Management System (ADAMS) Accession Number ML101160026).~~

~~This program shall establish the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analyses, provides a means for processing changes to instrumentation setpoints, and identifies setpoint methodologies to ensure instrumentation will function as required. The program shall ensure that testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verifies that instrumentation will function as required.~~

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~~a. The program shall list the Functions in the following specifications to which it applies:~~

~~1. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation - Operating [(Analog)] [(Digital)];"~~

5.5 Programs and Manuals

5.5.18 Setpoint Control Program (continued)

2. LCO 3.3.2, "Reactor Protective System (RPS) Instrumentation - Shutdown [(Analog)] [(Digital)];"
3. LCO [3.3.3, "Control Element Assembly Calculators (CEACs) (Digital)];"
4. [LCO 3.3.4,"Engineered Safety Features Actuation System (ESFAS) Instrumentation (Analog);"] [LCO 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Instrumentation (Digital);"]
5. [LCO 3.3.6, "Diesel Generator (DG) - Loss of Voltage Start (LOVS) (Analog);"] [LCO 3.3.7, "Diesel Generator (DG) - Loss of Voltage Start (LOVS) (Digital);"]
6. [LCO 3.3.7, "Containment Purge Isolation Signal (CPIS) (Analog);"] [LCO 3.3.8, "Containment Purge Isolation Signal (CPIS) (Digital);"]
7. [LCO 3.3.8, "Control Room Isolation Signal (CRIS) (Analog);"] [LCO 3.3.9, "Control Room Isolation Signal (CRIS) (Digital);"];
8. [LCO 3.3.9, "Chemical and Volume Control System (CVCS) Isolation Signal (Analog);"]
9. [LCO 3.3.10, "Fuel Handling Isolation Signal (FHIS) (Digital);"]
10. LCO 3.3.13, "[Logarithmic] Power Monitoring Channels [(Analog)."] [(Digital)."]

b. The program shall require the [Limiting Trip Setpoint (LTSP)], [Nominal Trip Setpoint (NTSP)], Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) of the Functions described in paragraph a. are calculated using the NRC approved setpoint methodology, as listed below. In addition, the program shall contain the value of the [LTSP], [NTSP], AV, AFT, and ALT (as applicable) for each Function described in paragraph a. and shall identify the setpoint methodology used to calculate these values.

---------- Reviewer's Note ----------
List the NRC safety evaluation report by letter, date, and ADAMS accession number (if available) that approved the setpoint methodologies.
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1. [Insert reference to NRC safety evaluation that approved the setpoint methodology.]

c. The program shall establish methods to ensure that Functions described in paragraph a. will function as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.

5.5 Programs and Manuals

5.5.18 Setpoint Control Program (continued)

d. ----------REVIEWER'S NOTE----------
A license amendment request to implement a Setpoint Control Program must list the instrument functions to which the program requirements of paragraph d. will be applied. Paragraph d. shall apply to all Functions in the Reactor Protection System and Engineered Safety Feature Actuation System specifications unless one or more of the following exclusions apply:

1. Manual actuation circuits, automatic actuation logic circuits or to instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.

2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.

3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.

The program shall identify the Functions described in paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The [LTSP] of these Functions are Limiting Safety System Settings. These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS and CHANNEL FUNCTIONAL TESTS that verify the [LTSP or NTSP].

1. The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified [LTSP or NTSP].

2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified [LTSP or NTSP] by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.

5.5 Programs and Manuals

~~5.5.18 Setpoint Control Program (continued)~~

~~3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.~~

~~4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the [LTSP or NTSP] at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the [LTSP or NTSP] provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).~~

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~~e. The program shall be specified in [insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference].]~~

~~[~~ 5.5.~~19~~

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6.8.4.q

Surveillance Frequency Control Program

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This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program. ~~]~~

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~~[~~ 5.5.~~20~~

17

6.8.4.s

Risk Informed Completion Time Program

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This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

5.5 Programs and Manuals

6.8.4.s 5.5.~~20~~ Risk Informed Completion Time Program (continued)

17

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~~REVIEWER'S NOTE~~

~~The Risk Informed Completion Time is only applicable in MODES supported by the licensee's PRA. Licensees applying the RICT Program to MODES other than MODES 1 and 2 must demonstrate that they have the capability to calculate a RICT in those MODES or that the risk indicated by their MODE 1 and 2 PRA model is bounding with respect to the lower MODE conditions.~~

6

MODES 1 and 2

b. A RICT may only be utilized in ~~MODE 1, 2 [, and 3, and MODE 4 while relying on steam generators for heat removal]~~;

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c. When a RICT is being used, any change to the plant configuration~~, as defined in NEI 06-09-A, Appendix A,~~ must be considered for the effect on the RICT.

within the scope of the Risk Informed Completion Time Program

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1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. ~~For emergent conditions,~~ If the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

e

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1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

d. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

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5.5 Programs and Manuals

5.5.20 Risk Informed Completion Time Program (continued)

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.] 1

[5.5.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

This Program provides controls for monitoring the condition of the neutron absorber used in the spent fuel pool storage racks to verify the Boron-10 areal density is consistent with the assumptions in the spent fuel pool criticality analysis. The program shall be in accordance with NEI 16-03-A, "Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools," Revision 0, May 2017 [, with the following exceptions:

1. . . .].] 2

JUSTIFICATION FOR DEVIATIONS
ITS 5.5, PROGRAMS AND MANUALS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. ISTS 5.5.2.b requires integrated leak test requirements for each system at least once per [18] months. ISTS 5.5.12.c requires total particulate sampling every 31 days. ISTS 5.5.17.d requires control room envelope pressure measurement testing relative to external adjacent areas to be performed at a frequency of [18] months on a STAGGERED TEST BASIS. St. Lucie Plant (PSL) controls periodic Frequencies for Surveillances in accordance with the Surveillance Frequency Control Program per Unit 1 CTS 6.8.4.o and Unit 2 CTS 6.8.4.q. Therefore, ITS 5.5.2.b, 5.5.10.c, and 5.5.15.d will be performed at a Frequency in accordance with the Surveillance Frequency Control Program.

4. ISTS 5.6.3, Post Accident Sampling, and ISTS 5.5.6, Pre-Stressed Concrete Containment Tendon Surveillance Program are not included in the ITS. The Post Accident Sampling requirements were deleted from the Unit 1 and Unit 2 CTS in March 27, 2001 by Amendments 174, and 114, respectively (ADAMS Accession No. ML011140017). Additionally, the containment design at PSL Unit 1 and Unit 2 does not include tendons. Subsequent Specifications are renumbered as a result of this deletion.

5. ISTS 5.5.7, Reactor Coolant Pump Flywheel Inspection Program, is changed to the PSL Unit 2 current licensing basis for flywheels composed of ASTM A-516-69 Grade 65 material and retains the ISTS requirement for flywheels composed of materials other than ASTM A-516-69 Grade 65. By letter dated October 9, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19282D338), as supplemented by letters dated April 30, 2020, and June 26, 2020 (ADAMS Accession Nos. ML20121A170 and ML20178A463, respectively), FPL submitted to the NRC, a license amendment request (LAR) to revise Technical Specification (TS) 6.8.4.o, “Reactor Coolant Pump Flywheel Inspection Program,” for PSL Unit 2. As discussed in Section 3.1 of this safety evaluation (SE), the licensee proposed to modify the St. Lucie Unit 2 Reactor Coolant Pump (RCP) Flywheel Inspection Program requirements to be consistent with the conclusions and limitations specified in the NRC’s May 21, 1997, SE of Topical Report SIR-94-080, Revision 1, “Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements” (ADAMS Accession No. ML20013C086). The proposed change to Unit 2 CTS 6.8.4.o. was approved in NRC SE, dated November 18, 2020 (ADAMS Accession No. ML20259A298). Unit 1 is incorporating an RCP Flywheel Inspection Program as a more restrictive programmatic requirement and, therefore, is including the same frequency requirement in Unit 1 ITS as Unit 2 ITS to allow RCP flywheel exchanges between Unit 1 and Unit 2.

6. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

7. The program details of the Explosive Gas and Storage Tank Radioactivity Monitoring Program are described in ISTS 5.5.11.a and 5.5.11.b (ITS 5.5.9.a and 5.5.9.b). Therefore, the sentence in the introductory paragraph that specifies a method to determine the explosive gas and storage tank radioactivity is not necessary. Additionally, the requirements specified in ISTS 5.5.11.c do not apply to PSL Unit 1 or Unit 2. UFSAR Chapter 15 states that a waste gas decay tank release is not a credible accident and that the dose will remain within the regulatory guidelines. This is also consistent with the CTS.

8. PSL Unit 1 and Unit 2 comply with Option B of 10 CFR 50, Appendix J. Therefore, the Option A and combined Option A and B provisions of ISTS 5.5.15 are not included in the ITS.

9. ISTS 5.5.16 is modified in ITS 5.5.14 to reference IEEE 450-2010 and Revision 3 of Regulatory Guide (RG) 1.129 instead of IEEE-450-2002 and Revision 2 of RG 1.129. RG 1.129, Revision 3 endorses the use of IEEE 450-2010 and eliminates the need for the exceptions specified in ISTS 5.5.16.a. Therefore, the exceptions of ISTS 5.5.16.a are not included in ITS 5.5.14. Section 5.4.2 of IEEE 450-2010 states, in part, that specific gravity readings are not recommended to be taken on a regular basis. PSL batteries are lead-calcium type batteries and therefore, specific gravities are not required to be obtained at each discharge test. Therefore, ISTS 5.5.16.b.5 is not included in ITS 5.5.14. Use of IEEE 450-2010 and RG 1.129, Revision 3 in the Battery Monitoring and Maintenance Program has been previously approved in Donald C. Cook Nuclear Plant Amendments 343 and 325 dated February 5, 2019 for Units 1 and 2, respectively (NRC ADAMS Accession No. ML18346A358).

10. PSL Unit 1 and Unit 2 CTS do not use a Setpoint Control Program for the control of instrumentation setpoints for automatic protective devices. Therefore, ISTS 5.5.18, Setpoint Control Program, is not included in the ITS. Subsequent Specifications are renumbered as a result of this deletion.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.5, PROGRAMS AND MANUALS

10 CFR 50.92 EVALUATION
FOR
MORE RESTRICTIVE CHANGE M01 – UNIT 1
LESS RESTRICTIVE CHANGE L02 – UNIT 2

Florida Power & Light Company (FPL) is converting the St. Lucie Plant (PSL) Unit 1 and Unit 2 Technical Specifications to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 5. The proposed change involves making the Unit 1 Current Technical Specifications (CTS) more restrictive and the Unit 2 CTS less restrictive. Below is the description of the change and the determination of No Significant Hazards Considerations for conversion to NUREG-1432.

Unit 1 CTS does not provide administrative controls for reactor RCP flywheel inspections. ITS 5.5.5, “Reactor Coolant Pump (RCP) Flywheel Inspection Program” is added to the Unit 1 ITS equivalent to the proposed Unit 2 program requirements. Unit 2 CTS requires administrative controls for RCP flywheel inspections. ITS 5.5.5 includes the same requirements and, in addition, includes program requirements for the Unit 2 CTS that were issued prior to Amendment No. 205, which are also consistent with the requirements specified in NUREG-1432, with a distinction on applicability of the two sets of requirements. The proposed program for both units will provide for the inspection of each RCP flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975 except for an RCP flywheel consisting of ASTM A 516 69 (i.e., SA-516) Grade 65 material. For an RCP flywheel composed of ASTM A 516-69 Grade 65 material, the inspection shall consist of either a 100% volumetric inspection of the upper flywheel over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheel at least once every 10 years. This changes the CTS by adding an additional administrative program to the Unit 1 CTS and restoring program requirements that were issued prior to Amendment No. 205 to the Unit 2 CTS, in addition to the current requirements.

The purpose of the RCP flywheel inspection program is to implement requirements to minimize the potential for failures of the flywheels of RCP motors in light-water-cooled power reactors. At PSL, RCP flywheels are interchanged between Unit 1 and Unit 2 during refurbishment and replacement. Between 2011 and 2017, the Unit 2 RCP flywheels, which consist of ASTM A-543 (three flywheels Class 1, Type B and one flywheel Grade 70) material, were exchanged with the Unit 1 RCP flywheels, which consist of ASTM A-516-69, Grade 65 material. Unit 2 CTS was modified to relax the RCP flywheel inspection requirements similar to the conclusions specified in the NRC Safety Evaluation of Topical Report SIR-94-080, Revision 1. The relaxed inspection requirements were approved in Unit 2 CTS Amendment No. 205, dated November 18, 2020 (ADAMS Accession No. ML20259A298). Previously, PSL Unit 2 RCP flywheels were inspected per NRC Regulatory Guide 1.14, “Reactor Coolant Pump Flywheel Integrity,” Revision 1 and the requirements were previously contained in the CTS pre-Amendment 205 requirements. NRC Regulatory Guide 1.14 describes a method acceptable to the NRC staff of implementing the RCP flywheel inspection program requirements and include a recommended inspection schedule. Since the PSL RCP flywheels consisting of ASTM A 543 material were previously inspected per Regulatory Guide 1.14, Revision 1, August 1975 and the requirements were previously approved for

use for Unit 2, the pre-Amendment 205 requirements are proposed to be restored in support of future RCP flywheel exchanges between Unit 1 and Unit 2.

FPL has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

Response: No.

The proposed license amendment would add RCP flywheel inspection program requirements to the Unit 1 Technical Specifications and revise the Unit 2 RCP flywheel inspection program by restoring previously NRC approved inspection requirements in addition to the current requirements. Adding an administrative program and restoring previously NRC approved inspection requirements neither affects the design of any plant structure, system or component (SSC), nor the manner in which the SSCs are operated and controlled. No changes are proposed to the facility or to any accident analysis assumptions, inputs or expected outcomes. Providing RCP flywheel inspection frequencies based on the flywheel metallurgical properties in the applicable Unit 2 administrative program and adding an equivalent administrative program to Unit 1 is acceptable because, for the flywheel metallurgical properties that satisfy the criteria specified in the NRC safety evaluation for topical report SIR-94-080, the increase inspection frequency has been determined acceptable by the NRC and approved for use at PSL Unit 2. Flywheels with other metallurgical properties will continue to be inspected using the methods and inspection schedule in accordance with the methods described in Regulatory Guide 1.14, “Reactor Coolant Pump Flywheel Integrity,” Revision 1. Compliance with the SIR-94-080 NRC safety evaluation and the NRC regulatory guide cannot adversely affect the likelihood or outcome of any design basis accident.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

Response: No

The proposed addition of RCP flywheel inspection program requirements to the Unit 1 Technical Specifications and revision of the Unit 2 RCP flywheel inspection program requirements does not affect RCP capability to provide adequate core cooling in the event of a power loss and thereby cannot create new inputs or assumptions associated with any accident analyses. The proposed change neither installs new plant equipment nor modifies the manner in which existing equipment is operated and controlled, and therefore cannot introduce new failure modes. Providing RCP flywheel inspection frequencies based on the flywheel metallurgical properties in the applicable Unit 2 administrative program and adding an equivalent administrative program to Unit 1 is acceptable because, for the flywheel metallurgical properties that satisfy the criteria specified in the NRC safety evaluation for topical report SIR-94-080, the increase inspection frequency has been determined acceptable by the NRC and approved for use

at PSL Unit 2. Flywheels with other metallurgical properties will continue to be inspected using the methods and inspection schedule in accordance with NRC Regulatory Guide 1.14. Compliance with the SIR-94-080 NRC safety evaluation and the NRC regulatory guide cannot create new or different kinds of accidents.

Therefore, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in the margin of safety?

Response: No.

The proposed addition of RCP flywheel inspection program requirements to the Unit 1 Technical Specifications and revision of the Unit 2 RCP flywheel inspection program requirements do not involve changes to any safety analyses, safety limits or limiting safety system settings. The proposed change does not adversely impact plant operating margins or the reliability of equipment credited in safety analyses. Providing RCP flywheel inspection frequencies based on the flywheel metallurgical properties in the applicable Unit 2 administrative program and adding an equivalent administrative program to Unit 1 is acceptable because, for the flywheel metallurgical properties that satisfy the criteria specified in the NRC safety evaluation for topical report SIR-94-080, the increase inspection frequency has been determined acceptable by the NRC and approved for use at PSL Unit 2. Flywheels with other metallurgical properties will continue to be inspected using the methods and inspection schedule in accordance with NRC Regulatory Guide 1.14. Compliance with the SIR-94-080 NRC safety evaluation and the NRC regulatory guide cannot result in a reduction in the margin of safety.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based upon the above analysis, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of no significant hazards consideration is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.5, PROGRAMS AND MANUALS

10 CFR 50.92 EVALUATION
FOR
REMOVE DETAIL CHANGE LA01

Florida Power & Light Company (FPL) is converting the St. Lucie Plant (PSL) Unit 1 and Unit 2 Technical Specifications to the Improved Technical Specifications (ITS) as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 5. The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of the change and the determination of No Significant Hazards Considerations for conversion to NUREG-1432. |R1

CTS contains periodic testing frequencies in administrative program requirements for: leakage testing of primary coolant sources outside containment; testing of the Shield Building Ventilation System filter heaters; total particulate concentration testing of the diesel generator fuel oil; measurement, and at designated locations, of the control room envelope (CRE) pressure relative to external areas adjacent to the CRE boundary during the pressurization mode of operation of the Control Room Emergency Ventilation System. ITS specifies the periodic Frequency for these tests as "In accordance with the Surveillance Frequency Control Program." This changes the CTS by moving the specified periodic Frequency for these tests to the Surveillance Frequency Control Program.

The purpose of these Surveillances is to assure that the necessary quality of systems and components is maintained. The removal of these details related to surveillance requirement frequencies from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The existing frequency is removed from Technical Specifications and placed under licensee control pursuant to the methodology described in Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1, dated April 2007 (ADAMS Accession No. ML071360456). The surveillance test requirements remain in the Technical Specifications. The control of changes to the surveillance frequencies is in accordance with the Surveillance Frequency Control Program, which is retained in the ITS. The Surveillance Frequency Control Program provides the necessary administrative controls to require that surveillances related to testing, calibration and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The proposed change to relocate periodic frequencies in the administrative controls section of Technical Specifications has been previously approved for Wolf Creek Generating Station Unit 1 in Amendment 227, dated April 8, 2021 (NRC ADAMS Accession No. ML21053A117), River Bend Station Unit 1 in Amendment 196, dated April 29, 2019 (NRC ADAMS Accession No. ML19066A008), and Grand Gulf Nuclear Station Unit 1 in Amendment 219, dated June 11, 2019 (NRC ADAMS Accession No. ML19094A799). PSL Unit 1 and Unit 2 adopted a Surveillance Frequency Control Program in Amendment Nos. 223 (Unit 1) and 173 (Unit 2) (ADAMS Accession No. ML15127A066). This change is acceptable because the testing frequencies will be adequately controlled in accordance with the Surveillance Frequency Control Program requirements retained in ITS, which ensure changes are properly evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.5, PROGRAMS AND MANUALS

FPL has reviewed the proposed no significant hazards consideration (NSHC) determination published in *Federal Register 74 FR 32000* dated July 6, 2009. FPL has concluded that the proposed NSHC presented in the Federal Register notice is applicable to the proposed relocation of the periodic testing frequencies specified herein.

As required by 10 CFR 50.91(a), the FPL analysis of the issue of no significant hazards consideration is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

Response: No.

The proposed change relocates the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the technical specifications for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

Response: No

No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in the margin of safety?

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the updated final safety analysis report and bases to technical specifications), since these are not affected by changes to the surveillance frequencies.

Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated ·surveillance frequency, FPL will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Rev. 1 in accordance with the Technical Specification Surveillance Frequency Control Program. NEI 04-10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications.”

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above analysis, FPL concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of no significant hazards consideration is justified.

ATTACHMENT 6

ITS 5.6 Section, Reporting Requirements

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

A01

ADMINISTRATIVE CONTROLS (continued)

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

See ITS 5.5

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate ACTIONS of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate ACTIONS to enter are those of the support system.

6.9 REPORTING REQUIREMENTS

~~ROUTINE REPORTS~~

5.6

~~6.9.1~~ In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC ~~in accordance with 10 CFR 50.4~~.

A02

~~STARTUP REPORT~~

~~6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following:~~

~~(1) receipt of an operating license,~~

~~(2) amendment of the license involving a planned increase in power level,~~

L01

~~(3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and~~

~~(4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.~~

ADMINISTRATIVE CONTROLS

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

L01

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS 1/

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Annual reports shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last

L02

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

~~isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~

L02

~~MONTHLY OPERATING REPORTS~~

~~6.9.1.6 Deleted~~

A01

A01

ADMINISTRATIVE CONTROLS

~~ANNUAL~~ RADIOACTIVE EFFLUENT RELEASE REPORT*

A01

5.6.2 ~~6.9.1.7~~ The ~~Annual~~ Radioactive Effluent Release Report covering the operation of the unit ~~during the previous 12 months of operation~~ shall be submitted ~~within 60 days after January 1 of each year~~. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

prior to May 1 of each year in accordance with 10 CFR 50.36a

L03

5.6.2 Note * A single submittal may be made for a multiple unit station. The submittal shall ~~should~~ combine ~~those~~ sections ~~that are~~ common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

A01

A01

ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT**

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

M01

5.6.1

~~6.9.1.8~~ The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted ~~before May 1~~ by May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in ~~(1)~~ the ~~ODCM~~ in Offsite Dose Calculation Manual (ODCM) and ~~(2)~~ Sections IV.B.2, IV.B.3, and IV.C ~~of~~ Appendix I ~~to~~ 10 CFR Part 50.

L03

5.6.1 Note

** A single submittal may be made for a multiple unit station.

The submittal should combine sections common to all units at the station.

A01

ADMINISTRATIVE CONTROLS

~~ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT~~ ~~(continued)~~

~~6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.~~

~~6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.~~

LA01

5.6.3 **~~6.9.1.11~~ CORE OPERATING LIMITS REPORT (COLR)**

5.6.3.a a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

5.6.3.a
Document List
ITS Specifications

Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater Than 200°F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200°F
Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Full Length CEA Position – Misalignment > 15 inches
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factor – F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations – Boron Concentration

5.6.3.b b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, ~~as~~ specifically those described in the following documents, approved Revisions and Supplements as specified in the COLR.

5.6.3.b
Document List

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary)

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. XN-75-27(A) [also issued as XN-NF-75-27(A)], "Exxon Nuclear Neutronic(s) Design Methods for Pressurized Water Reactors"

4. ~~DELETED~~

A01

5. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations"

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

5.6.3.b

6. ~~DELETED~~ A01

7. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing"

8. ~~DELETED~~ A01

9. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors"

10. XN-NF-621(P)(A), " Exxon Nuclear DNB Correlation for PWR Fuel Designs"

11. ~~DELETED~~ A01

12. XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup"

13. ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU"

14. XN-NF-85-92 (P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results"

15. ~~DELETED~~ A01

16. ~~DELETED~~ A01

17. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Design"

18. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel"

19. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results"

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

5.6.3.b

20. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors"

21. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Revision 1, as supplemented by ANP-3000(P), “St. Lucie Unit 1 EPU - Information to Support License Amendment Request,” Revision 0.

22. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” Revision 0, as supplemented by ANP-3000(P), “St. Lucie Unit 1 EPU - Information to Support License Amendment Request,” Revision 0.

23. EMF-2103(P)(A), “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” Revision 0, as supplemented by ANP-2903(P), “St. Lucie Nuclear Plant Unit 1 EPU Cycle Realistic Large Break LOCA summary Report with Zr-4 Fuel Cladding,” Revision 1.

24. BAW-10240(P)(A) Revision 0, “Incorporation of M5 Properties in Framatome ANP Approved Methods.”

25. WCAP-16045-P-A, Revision 0, “Qualification of the Two -Dimensional Transport Code PARAGON,” August 2004.

A01

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

assuming operation up to RATED THERMAL POWER

A01

5.6.3.c c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

5.6.3.d d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.5 ### STEAM GENERATOR TUBE INSPECTION REPORT

~~6.9.1.12~~ A report shall be submitted within 180 days after the initial entry into ~~HOT SHUTDOWN~~ MODE 4 following completion of an inspection performed in accordance with Specification ~~6.8.4.l~~ 5.5.6, Steam Generator (SG) Program. The report shall include:

5.6.5 a a. The scope of inspections performed on each SG,

Add proposed ITS 5.6.5.b

M02

5.6.5 c b. ~~Active~~ degradation mechanisms found,

A03

5.6.5 c.1 c. Nondestructive examination techniques utilized for each degradation mechanism,

and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

5.6.5 c.2 d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

5.6.5 c.4 e. Number of tubes plugged during the inspection outage for each ~~active~~ degradation mechanism,

found

Add proposed ITS 5.6.5.d

M02

The

5.6.5.e f. ~~Total~~ number and percentage of tubes plugged to date,

and

5.6.5.c.3 g. The results of condition monitoring~~, including the results of tube pulls and in-situ testing, and~~

5.6.5.e h. The effective plugging percentage ~~for all plugging~~ in each SG.

Add proposed ITS 5.6.5.f

M02

~~SPECIAL REPORTS~~

~~6.9.2~~ ~~Special reports shall be submitted to the NRC within the time period specified for~~ each report.

A01

including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment;

A03

ADMINISTRATIVE CONTROLS

~~6.10 DELETED~~

~~DELETED~~

~~TABLE 3.3-6 (Continued)~~

~~TABLE NOTATION~~

ACTION 12 - DELETED

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

See ITS 3.3.9

5.6.4
ITS 3.3.9
Action F

~~1)~~ ~~Initiate the~~ preplanned alternate method of monitoring the appropriate parameter(s),and

R1

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation,"

~~2)~~ Prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to Specification6.9.2~~ within 14 days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ to OPERABLE status.

A01

instrumentation channels of the Function

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

~~TABLE 3.3-11~~ (continued)

~~ACTION STATEMENTS~~

ACTION 1 - With the number of OPERABLE channels less than the Total No. of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 2 - With position indication inoperable, restore the inoperable indicator to OPERABLE status or close the associated PORV block valve and remove power from its operator within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

See ITS 3.3.9

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information once per shift to determine valve position.

ACTION 4 - With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to the specification 6.9.2~~ within ~~30~~ days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ to OPERABLE status.

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation,"

5.6.4 preplanned alternate method of monitoring

instrumentation channels of the Function

14

M03

ACTION 5 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

See ITS 3.3.9

5.6.4
ITS 3.3.9
Action F

1. ~~Initiate an~~ alternate method of monitoring the reactor vessel inventory; and

R1

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation,"

2. ~~2.~~ Prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to Specification 6.9.2~~ within ~~30~~ days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ to OPERABLE status; and

14

M03

instrumentation channels of the Function

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

ACTION 6 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

See ITS 3.3.9

ACTION 7 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 72 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

A01

ADMINISTRATIVE CONTROLS

6.9 REPORTING REQUIREMENTS

~~ROUTINE REPORTS~~

5.6

~~6.9.1~~ In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC. in accordance with 10 CFR 50.4 A02

~~STARTUP REPORT~~

~~6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier; and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.~~

~~6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

L01

~~6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

ADMINISTRATIVE CONTROLS

~~ANNUAL REPORTS~~ ~~1/~~

~~6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.~~

~~6.9.1.5 Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.~~

L02

~~MONTHLY OPERATING REPORTS~~

A01

~~6.9.1.6 Deleted~~

~~1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.~~

L02

A01

ADMINISTRATIVE CONTROLS

~~ANNUAL~~ RADIOACTIVE EFFLUENT RELEASE REPORT*

A01

5.6.2 ~~6.9.1.7~~ The ~~Annual~~ Radioactive Effluent Release Report covering the operation of the unit ~~during the previous 12 months of operation~~ shall be submitted ~~within 60 days after January 1 of each year~~. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

prior to May 1 of each year in accordance with 10 CFR 50.36a

L03

5.6.2 Note * A single submittal may be made for a multiple unit station. The submittal shall ~~should~~ combine ~~those~~ sections ~~that are~~ common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

A01

ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT**

5.6.1 ~~6.9.1.8~~ The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted ~~before May 1~~ by May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in ~~(1)~~ the ~~ODCM~~ Offsite Dose Calculation Manual (ODCM) and ~~(2)~~ Sections IV.B.2, IV.B.3, and IV.C ~~of~~ Appendix I ~~to~~ 10 CFR Part 50.

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in

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

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** A single submittal may be made for a multiple unit station.

The submittal should combine sections common to all units at the station.

A01

A01

ADMINISTRATIVE CONTROLS

~~ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT~~ ~~(continued)~~

~~6.9.1.9~~ ~~At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.~~

~~6.9.1.10~~ ~~At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.~~

LA01

5.6.3 ~~6.9.1.11~~ CORE OPERATING LIMITS REPORT (COLR)

5.6.3.a a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

5.6.3.a
Document List
ITS Specifications

Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater than 200°F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200°F
Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Movable Control Assemblies – CEA Position
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factors – F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations – Boron Concentration

5.6.3.b b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, ~~as~~ specifically those described in the following documents or any approved Revisions and Supplements thereto:

5.6.3.b
Document List

1. WCAP-11596-P-A, “Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” June 1988 (Westinghouse Proprietary).

2. NF-TR-95-01, “Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants,” Florida Power & Light Company, January 1995.

A01

3. ~~DELETED~~

A01

4. ~~DELETED~~

5. CENPD-275-P, Revision 1-P-A, “C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers,” May 1988, & Revision 1-P Supplement 1-P-A, April 1999.

A01

6. ~~DELETED~~

A01

ADMINISTRATIVE CONTROLS (Continued)

CORE OPERATING LIMITS REPORT (COLR) (Continued)

5.6.3.b

b. (Continued)

7. ~~DELETED~~ A01

8. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 1: CE Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for St. Lucie Unit 1," December 1979.

9. ~~DELETED~~ A01

10. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 3: CE Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for St. Lucie Unit 1," February 1980.

11. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981.

12. Letter, J.W. Miller (NRC) to J.R. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval of CEN-123(F)-P (three parts) and CEN-191(B)-P).

13. ~~DELETED~~ A01

14. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), Docket No. 50-389, "St. Lucie Unit 2 – Change to Technical Specification Bases Sections '2.1.1 Reactor Core' and '3/4.2.5 DNB Parameters' (TAC No. M87722)," March 14, 1994 (Approval of CEN-371(F)-P).

15. ~~DELETED~~ A01

16. ~~DELETED~~ A01

17. ~~DELETED~~ A01

18. ~~DELETED~~ A01

ADMINISTRATIVE CONTROLS (Continued)

CORE OPERATING LIMITS REPORT (COLR) (Continued)

5.6.3.b

b. (Continued)

19. CENPD-225-P-A, “Fuel and Poison Rod Bowing,” June 1983.

20. CENPD-139-P-A, “C-E Fuel Evaluation Model Topical Report,” July 1974.

21. CEN-161(B)-P-A, “Improvements to Fuel Evaluation Model,” August 1989.

22. CEN-161(B)-P, Supplement 1-PA, “Improvements to Fuel Evaluation Model,” January 1992.

23. CENPD-132, Supplement 3-P-A, “Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS,” June 1985.

24. CENPD-133, Supplement 5-A, “CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis,” June 1985.

25. CENPD-134, Supplement 2-A, “COMPERC-II, a Program for Emergency Refill-Reflood of the Core,” June 1985.

26. CENPD-135-P, Supplement 5, “STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program,” April 1977.

27. Letter, R.L. Baer (NRC) to A.E. Scherer (CE), “Evaluation of Topical Report CENPD-135, Supplement #5,” September 6, 1978.

28. CENPD-137, Supplement 1-P, “Calculative Methods for the C-E Small Break LOCA Evaluation Model,” January 1977.

29. CENPD-133, Supplement 3-P, “CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident,” January 1977.

30. Letter, K. Kneil (NRC) to A.E. Scherer (CE), “Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P,” September 27, 1977.

31. CENPD-138, Supplement 2-P, “PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup,” January 1977.

32. Letter, C. Aniel (NRC) to A.E. Scherer (CE), “Evaluation of Topical Report CENPD-138, Supplement 2-P,” April 10, 1978.

33. Letter, W.H. Bohlke (FPL) to Document Control Desk (NRC), “St. Lucie Unit 2, Docket No. 50-389, Proposed License Amendment, MTC Change from –27 pcm to –30 pcm,” L-91-325, December 17, 1991.

ADMINISTRATIVE CONTROLS (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

5.6.3.b

b. (continued)

34. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), "St. Lucie Unit 2 – Issuance of Amendment Re: Moderator Temperature Coefficient (TAC No. M82517)," July 15, 1992.

35. Letter, J.W. Williams, Jr. (FPL) to D.G. Eisenhut (NRC), "St. Lucie Unit No. 2, Docket No. 50-389, Proposed License Amendment, Cycle 2 Reload," L-84-148, June 4, 1984.

36. Letter, J.R. Miller (NRC) to J.W. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval to Methodology contained in L-84-148).

37. ~~DELETED~~

38. ~~DELETED~~

39. ~~DELETED~~

40. ~~DELETED~~

41. ~~DELETED~~

A01

42. CEN-348(B)-P-A, Supplement 1-P-A, "Extended Statistical Combination of Uncertainties," January 1997.

43. CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

44. ~~DELETED~~

45. ~~DELETED~~

A01

ADMINISTRATIVE CONTROLS (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

5.6.3.b

b. (continued)

46. ~~DELETED~~

47. ~~DELETED~~

A01

48. CEN-396(L)-P, “Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/KG for St. Lucie Unit 2,” November 1989 (NRC SER dated October 18, 1991, Letter J.A. Norris (NRC) to J.H. Goldberg (FPL), TAC No. 75947).

49. CENPD-269-P, Rev. 1-P, “Extended Burnup Operation of Combustion Engineering PWR Fuel,” July 1984.

50. CEN-289(A)-P, “Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2,” December 1984 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

51. CENPD-137, Supplement 2-P-A, “Calculative Methods for the ABB CE Small Break LOCA Evaluation Model,” April 1998.

52. CENPD-140-A, “Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis,” June 1976.

53. ~~DELETED~~

54. ~~DELETED~~

A01

55. CENPD-387-P-A, Revision 000, “ABB Critical Heat Flux Correlations for PWR Fuel,” May 2000.

56. CENPD-132, Supplement 4-P-A, “Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

57. CENPD-137, Supplement 2-P-A, “Calculative Methods for the ABB CE Small Break LOCA Evaluation Model,” April 1998.

58. CENPD-404-P-A, Rev. 0, “Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs,” November 2001.

59. WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985.

60. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification,” February 1994.

ADMINISTRATIVE CONTROLS (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

5.6.3.b

b. (continued)

61. WCAP-11397-P-A, (Proprietary), 'Revised Thermal Design Procedure," April 1989.

62. WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

63. WCAP-14565-P-A, Addendum 1, "Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003.

64. Letter, W. Jefferson Jr. (FPL) to Document Control Desk (USNRC), “St. Lucie Unit 2 Docket No. 50-389: Proposed License Amendment WCAP-9272 Reload Methodology and Implementing 30% Steam Generator Tube Plugging Limit,” L-2003-276, December 2003 (NRC SER dated January 31, 2005, Letter B.T. Moroney (NRC) to J.A. Stall (FPL), TAC No. MC1566).

65. WCAP-14882-P-A, Rev. 0, “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses”, April 1999.

66. WCAP-7908-A, Rev. 0, “FACTRAN-A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod”, December 1989.

67. WCAP-7979-P-A, Rev. 0, “TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code”, January 1975.

68. WCAP-7588, Rev. 1-A, “An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Special Kinetics Methods”, January 1975.

69. WCAP-16045-P-A, Revision 0, “Qualification of the Two-Dimensional Transport Code PARAGON,” August 2004.

70. EMF-96-029(P)(A), Volumes 1 and 2, “Reactor Analysis System for PWRs, Volume 1 Methodology Description, Volume 2 Benchmarking Results,” Siemens Power Corporation, January 1997.

71. XN-NF-78-44 (NP)(A), “A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors,” Exxon Nuclear Company, Inc.,” October 1983.

72. XN-75-27(A) and Supplements 1 through 5, “Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors,” Exxon Nuclear Company, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P).

73. XN-NF-82-06 (P)(A), Rev. 1 and Supplements 2, 4, and 5, “Qualification of Exxon Nuclear Fuel for Extended Burnup,” Exxon Nuclear Company, Inc., October 1986.

ADMINISTRATIVE CONTROLS (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

5.6.3.b

b. (continued)

74. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, Inc., November 1986.

75. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, December 1991.

76. EMF-92-116(P)(A), Rev. 0, "Generic Mechanical Design Criteria for PWR Fuel Design," Siemens Power Corporation, February, 1999.

77. BAW-10240(P)(A), Rev.0, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," Framatome ANP, Inc., May 2004.

78. XN-NF-82-21(P)(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, September 1983.

79. EMF-92-153(P)(A), Revision 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," January 2005.

80. EMF-1961(P)(A), Revision 0, "Statistical/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, July 2000.

81. EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP, Inc., May 2004.

82. XN-75-32(P)(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bowing," October 1983.

83. BAW-10231P-A Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004.

84. EMF-2103(P)(A) Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.

85. EMF-2328 (P)(A) Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001.

assuming operation up to RATED THERMAL POWER

A01

5.6.3.c

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

5.6.3.d

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle on the NRC.

A01

ADMINISTRATIVE CONTROLS (continued)

SPECIAL REPORTS

STEAM GENERATOR TUBE INSPECTION REPORT

MODE 4

5.6.5 ~~6.9.1.12~~ A report shall be submitted within 180 days after the initial entry into ~~HOT SHUTDOWN~~ following completion of an inspection of the replacement SGs performed in accordance with Specification ~~6.8.4.l.1~~. The report shall include:

5.5.6

5.6.5 a a. The scope of inspections performed on each SG,

Add proposed ITS 5.6.5.b

M02

5.6.5 c b. ~~Active~~ degradation mechanisms found,

5.6.5 c.1 c. Nondestructive examination techniques utilized for each degradation mechanism,

A03

5.6.5 c.2 d. Location, orientation (if linear), and measured sizes (if available) of service induced indications, and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

5.6.5 c.4 e. Number of tubes plugged during the inspection outage for each active degradation mechanism; found

Add proposed ITS 5.6.5.d

M02

The

5.6.5.e f. ~~Total~~ number and percentage of tubes plugged to date,

and

5.6.5.c.3 g. The results of condition monitoring~~, including the results of tube pulls and in-situ testing,~~

5.6.5.e h. The effective plugging percentage ~~for all plugging~~ in each SG.

~~6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.~~

A01

Add proposed ITS 5.6.5.f

M02

~~**6.10 DELETED**~~

including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment;

A03

~~TABLE 3.3-6 (Continued)~~

~~ACTION STATEMENTS~~

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

See ITS 3.3.8

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

See ITS 3.4.15

ACTION 24 - DELETED

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

See ITS 3.3.6

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

See ITS 3.3.7

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

See ITS 3.3.9

5.6.4
ITS 3.3.9
Action F

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation,"

1) ~~Initiate the~~ preplanned alternate method of monitoring the appropriate parameter(s), and

R1

2) Prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to Specification 6.9.2~~ within 14 days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ to OPERABLE status.

A01

instrumentation channels of the Function

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

See ITS 3.3.9

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a.* With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

b.* With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation,"

c.** With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to Specification 6.9.2~~ within ~~30~~ days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ to OPERABLE status.

5.6.4 preplanned alternate method of monitoring

14

M03

instrumentation channels of the Function

See ITS 3.3.9

d.** With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

5.6.4
ITS 3.3.9
Action F

When a report is required by Condition B or F of LCO 3.3.9, "Post Accident Monitoring (PAM) Instrumentation,"

1. ~~Initiate an~~ alternate method of monitoring the reactor vessel inventory; and | R1

2. Prepare and submit a ~~Special~~ Report ~~to the Commission pursuant to Specification 6.9.2~~ within ~~30~~ days following the event outlining the ~~action taken~~, the cause of the inoperability and the plans and schedule for restoring the ~~system~~ to OPERABLE status, ~~and~~

instrumentation channels of the Function

14

M03

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

See ITS 3.3.9

* Action statements do not apply to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

** Action statements apply only to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

DISCUSSION OF CHANGES
ITS 5.6, REPORTING REQUIREMENTS

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2, Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1432, Rev. 5.0, "Standard Technical Specifications-Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 6.9.1 specifies in addition to the applicable reporting requirements of 10 CFR, the reports that must be submitted to the NRC. ITS 5.6 requires that reports be submitted in accordance with 10 CFR 50.4. This changes the CTS by adding the 10 CFR 50.4 requirement.

10 CFR 50.4 provides distribution requirements for written communications to the NRC. This change is acceptable because the 10 CFR 50.4 distribution requirements already apply to each report to be submitted to the NRC. This change is designated as administrative because it does not result in technical changes to the CTS.

A03 CTS 6.9.1.12 provides requirements for the Steam Generator (SG) Tube Inspection Report and lists items to include in the report, including the requirement to provide location, orientation, and measured sizes of each degradation mechanism found, and the results of condition monitoring including the results of tube pulls and in situ testing. ITS 5.6.5 also provides requirements for the SG Tube Inspection Report and a list of items to include in the report the requirement to provide location, orientation, and measured sizes of each degradation mechanism found, and the results of condition monitoring. ITS 5.6.5.c.2 and c.3 include additional detail on what to include in the report. ITS 5.6.5.c.2 clarifies that for any tube degradation mechanisms found, the report must include voltage response for each indication and, for tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported. ITS 5.6.5.c.3 clarifies that the description of the condition monitoring assessment and results must include the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment. The purpose of the report is to advise the NRC of results of SG tube inspections performed in accordance with the plant's steam generator program, which monitors and manages tube degradation and degradation precursors. These changes are consistent with the ISTS and approved for use by the NRC as specified in the safety evaluation associated with TSTF-577-A, Revision 1, dated April 14, 2021 (NRC ADAMS Accession No. ML21098A188). The NRC staff reviewed the proposed changes to the ISTS SG Tube Inspection Report and determined that they are acceptable because they will provide additional detailed information to allow the NRC staff to better understand the overall condition of the SGs. This change solely provides clarification detail on reporting requirements currently listed in the CTS. These clarifications to what must be provided in the SG Tube Inspection Report for the CTS items specified herein do

not result in a technical change to the CTS and, therefore, the change is designated as administrative.

MORE RESTRICTIVE CHANGES

M01 The second paragraph of ITS 5.6.1 includes details required to be included in the Annual Radiological Environmental Operating Report. CTS 6.9.1.8 does not contain this level of detail. This changes the CTS by requiring additional detail to be included in the Annual Radiological Environmental Operating Report.

The purpose of the second paragraph of ITS 5.6.1 is to specify details to be included in the Annual Radiological Environmental Operating Report. This change is acceptable because the content requirements are consistent with the objectives outlined in the Offsite Dose Calculation Manual. This change is designated more restrictive because it adds new reporting requirements to the Technical Specifications.

M02 CTS 6.9.1.12 provides requirements for the SG Tube Inspection Report and lists items to include in the report. ITS 5.6.5 also provides requirements for the SG Tube Inspection Report and a list of items to include in the report. ITS 5.6.5 additionally requires the report to include: 1) the nondestructive examination techniques utilized for tubes with increased degradation susceptibility (ITS 5.6.5.b); 2) an analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results (ITS 5.6.5.d); and 3) the results of any SG secondary side inspections (ITS 5.6.5.f). This changes the CTS by requiring additional items to be included in the SG Tube Inspection Report that is submitted to the NRC.

The purpose of the report is to advise the NRC of results of SG tube inspections performed in accordance with the plant's steam generator program, which monitors and manages tube degradation and degradation precursors. These changes are consistent with the ISTS and approved for use by the NRC as specified in the safety evaluation associated with TSTF-577-A, Revision 1, dated April 14, 2021 (NRC ADAMS Accession No. ML21098A188). The NRC staff reviewed the proposed changes to the ISTS SG Tube Inspection Report and determined that they are acceptable because they will provide additional detailed information to allow the NRC staff to better understand the overall condition of the SGs. The proposed change is designated more restrictive because additional items are required in the ITS to be included in the SG Tube Inspection Report than required in the CTS.

M03 Unit 1 CTS Table 3.3-11 Actions 4 and 5, and Unit 2 CTS 3.3.3.6 Actions c and d contain a requirement that becomes applicable if the applicable channels are not restored to operable status within the allowed time. These actions require submittal of a Special Report to the Commission within 30 days outlining the action taken, the cause of the inoperability and the plans and schedule to restore the associated parameter to OPERABLE status. ITS Specification 5.6.4 requires the same report to be submitted within 14 days. This changes the CTS by

requiring a more timely NRC report to be submitted when channel(s) are not restored to OPERABLE status within the specified Completion Time.

The purpose of the CTS (and ITS) Actions is to advise the NRC regarding the failure to restore the required post accident monitoring channels to OPERABLE status in the allowed time. The CTS requirement allows the NRC report to be submitted within 30 days. The proposed change is acceptable because it provides a more timely report to the NRC. The proposed change does not impose an undue burden on plant personnel nor does the proposed change adversely affect the safe operation of the plant. The proposed change is designated more restrictive because a more timely report is required in the ITS than in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 6.9.1.9 and CTS 6.9.1.10 require reporting for estimates of actual population within 10 miles of the plant every 5 years, and within 50 miles of the plant every 10 years, respectively. ITS 5.6 does not contain this requirement. This changes the CTS by removing the population reporting requirements from the CTS and relocating them to the UFSAR, which contains these population reporting requirements.

Unit 1 UFSAR, Section 2.1.3.9 and Unit 2 UFSAR, Section 2.1.3.3 provide requirements for periodic update of population data. The UFSAR states, in part, that commencing in April 1993 and at least every 5 years thereafter, FPL will prepare and submit to the NRC an estimate of the actual population within 10 miles of the plant. In addition, commencing in April 1993 and at least every 10 years thereafter, FPL will prepare and submit to the NRC an estimate of the actual population within 50 miles of the plant.

The removal of these details for making reports from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. Also, this change is acceptable because these types of procedural details are adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because reporting requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L01 *(Category 9 – Deletion of Reporting Requirements)* CTS 6.9.1.1, CTS 6.9.1.2, and CTS 6.9.1.3 contain requirements for submitting a report of plant startup and power escalation testing following receipt of an operating license; amendments to

the license involving planned increase in power level; installation of fuel that has a different design or has been manufactured by a different fuel supplier; and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit. The ITS does not contain such reporting requirements. This changes the CTS by deleting the requirements of CTS 6.9.1.1, CTS 6.9.1.2, and CTS 6.9.1.3.

The purpose of CTS 6.9.1.1, CTS 6.9.1.2 and CTS 6.9.1.3 is to provide a summary of plant startup and power escalation testing following the four specified conditions as verification that the unit operated as expected. This change is acceptable because the regulations provide adequate reporting requirements. If there were any unit conditions outside the expected parameters during unit startup, they would be reported to the NRC if they met the reporting requirements in the regulations. Otherwise, the reports would document that the unit operated as expected and already approved by the NRC, as required by regulations. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L02 *(Category 9 - Deletion of Reporting Requirements)* CTS 6.9.1.4 requires the annual reports of CTS 6.9.1.5 regarding any instances when the specific activity limit of Specification 3.4.8 is exceeded, to be submitted prior to March 31 of each year. ITS 5.6 does not contain any requirements for such a report. This changes the CTS by not including the requirements for the annual reporting of instances when the Technical Specification specific activity limit for the primary coolant is exceeded.

The purpose of CTS 6.9.1.4 and CTS 6.9.1.5 is to specify the requirements for submitting information regarding any instances when the Technical Specification specific activity of Specification 3.4.8 is exceeded in an annual report. This change is acceptable because the regulations provide adequate details of reporting requirements, and the reporting of exceeding the 1-131 specific activity limit does not affect continued plant operation. Operations or conditions prohibited by the plant's Technical Specifications are required to be reported in accordance with 10 CFR 50.73. Subsequent reports would be provided if necessary, without requiring a specific annual report. This change is designated as less restrictive because the reports that would be submitted under the CTS will not be required under the ITS.

L03 *(Category 1 – Relaxation of LCO Requirements)* CTS 6.9.1.7 requires the Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year. ITS 5.6.1 requires the Radioactive Effluent Release Report to be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. CTS 6.9.1.8 requires the Annual Radiological Environmental Operating Report to be submitted before May 1 of each year. ITS 5.6.1 requires the Annual Radiological Environmental Operating Report to be submitted by May 15 of each year. This changes the CTS by allowing an additional time to submit these reports each year.

The purpose of the due date for submitting the Radioactive Effluent Release Report and Annual Radiological Environmental Operating Report is to ensure that the reports are provided in a reasonable period of time to the NRC for

review. This change is acceptable because the reports are still required to be submitted in a reasonable time frame. Given that the reports are still required to be provided to the NRC on or before May 1 or May 15, respectively, and cover the previous calendar year, report completion and submittal is clearly not necessary to assure operation in a safe manner for the interval between March 1 and April 30, and May 1 and May 15, respectively. Additionally, there is no requirement for the NRC to approve the reports. This change is designated as less restrictive because it allows more time to prepare and submit the reports to the NRC.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

6.9.1.8

5.6.1 Annual Radiological Environmental Operating Report

----------NOTE----------

6.9.1.8 Footnote **

~~[~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~]~~

2

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

DOC M01

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements ~~[in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]~~. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

2

6.9.1.7

5.6.2 Radiological Effluent Release Report

----------NOTE----------

6.9.1.7 Footnote *

~~[~~A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~]~~

2

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6 Reporting Requirements

6.9.1.11 **5.6.3 CORE OPERATING LIMITS REPORT**

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:, including approved revisions and supplements, as specified thereto (1)

INSERT 1 → ~~[The individual specifications that address core operating limits must be referenced here.]~~ (2)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

~~----------REVIEWER'S NOTE----------~~
~~Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the COLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.~~
~~----------~~ (3)

INSERT 2 → ~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~ (2)

c. The core operating limits shall be determined assuming operation up to RATED THERMAL POWER such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

~~5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT~~

~~a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:~~

~~[The individual specifications that address RCS pressure and temperature limits must be referenced here.]~~ (4)

2

INSERT 1

LCO 3.1.1	SHUTDOWN MARGIN (SDM)
LCO 3.1.3	Moderator Temperature Coefficient (MTC)
LCO 3.1.4	Control Element Assembly (CEA) Alignment
LCO 3.1.6	Regulating Control Element Assembly (CEA) Insertion Limits
LCO 3.2.1	Linear Heat Rate (LHR)
LCO 3.2.2	Total Integrated Radial Peaking Factor (F_r^T)
LCO 3.2.4	AXIAL SHAPE INDEX (ASI)
LCO 3.4.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
LCO 3.9.1	Boron Concentration

2

INSERT 2

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary)

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. XN-75-27(A) [also issued as XN NF 75 27(A)], "Exxon Nuclear Neutronic(s) Design Methods for Pressurized Water Reactors"

4. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations

5. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing"

6. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors"

7. XN-NF-621(P)(A), " Exxon Nuclear DNB Correlation for PWR Fuel Designs"

8. XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup"

9. ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU"

10. XN-NF-85-92 (P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results"

11. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Design"

12. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel"

13. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results"

14. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors"

15. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Revision 1, as supplemented by ANP-3000(P), “St. Lucie Unit 1 EPU - Information to Support License Amendment Request,” Revision 0.

16. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” Revision 0, as supplemented by ANP-3000(P), “St. Lucie Unit 1 EPU - Information to Support License Amendment Request,” Revision 0.

17. EMF-2103(P)(A), “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” Revision 0, as supplemented by ANP-2903(P), “St. Lucie Nuclear Plant Unit 1

2

INSERT 2

EPU Cycle Realistic Large Break LOCA summary Report with Zr-4 Fuel Cladding," Revision 1.

18. BAW-10240(P)(A) Revision 0, "Incorporation of M5 Properties in Framatome ANP Approved Methods."

19. WCAP-16045-P-A, Revision 0, "Qualification of the Two -Dimensional Transport Code PARAGON," August 2004.

5.6 Reporting Requirements

5.6.4 RCS Pressure and Temperature Limits Report (continued)

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

----------REVIEWER'S NOTE----------
Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the PTLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.

[Identify the NRC staff approval document by date.]

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

----------REVIEWER'S NOTE----------
The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

4

5.6 Reporting Requirements

~~5.6.4 RCS Pressure and Temperature Limits Report (continued)~~

~~7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~

4

1

Table 3.3-6, Action 15
Table 3.3-11, Actions 4 and 5
DOC M03

5.6.~~5~~ Post Accident Monitoring Report

4

9

When a report is required by Condition B or F of LCO 3.3.~~[11]~~, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

~~5.6.6 Tendon Surveillance Report~~

~~[Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]~~

5

1

6.9.1.12

5.6.~~7~~ Steam Generator Tube Inspection Report

5

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification ~~5.5.9~~ 5.5.6, "Steam Generator (SG) Program." The report shall include:

6.9.1.12.a

a. The scope of inspections performed on each SG;

DOC M02

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

6.9.1.12.c

1. The nondestructive examination techniques utilized;

5.6 Reporting Requirements

6.9.1.12 5.6.~~7~~ Steam Generator Tube Inspection Report (continued) 1

5

6.9.1.12.d DOC A03

2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

6.9.1.12.g DOC A03

3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment;

6.9.1.12.e

4. The number of tubes plugged ~~[or repaired]~~ during the inspection outage~~;~~. and

~~[5. The repair methods utilized and the number of tubes repaired by each repair method.]~~ 2

DOC M02

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

6.9.1.12.f 6.9.1.12.h

e. The number and percentage of tubes plugged ~~[or repaired]~~ to date, and the effective plugging percentage in each SG; 2

DOC M02

f. The results of any SG secondary side inspections~~;~~. and

~~[g. Insert any plant-specific reporting requirements, if applicable.]~~ 2

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

6.9.1.8

5.6.1 Annual Radiological Environmental Operating Report

----------NOTE----------

6.9.1.8 Footnote **

~~[~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~]~~

2

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

DOC M01

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements ~~[in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]~~. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

2

6.9.1.7

5.6.2 Radiological Effluent Release Report

----------NOTE----------

6.9.1.7 Footnote *

~~[~~A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~]~~

2

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6 Reporting Requirements

6.9.1.11 **5.6.3 CORE OPERATING LIMITS REPORT**

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:, including approved revisions and supplements, as specified thereto (1)

INSERT 1 → ~~[The individual specifications that address core operating limits must be referenced here.]~~ (2)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

~~----------REVIEWER'S NOTE----------~~
~~Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the COLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.~~
~~----------~~ (3)

INSERT 2 → ~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~ (2)

c. The core operating limits shall be determined assuming operation up to RATED THERMAL POWER such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

~~5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT~~

~~a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:~~

~~[The individual specifications that address RCS pressure and temperature limits must be referenced here.]~~ (4)

2

INSERT 1

LCO 3.1.1	SHUTDOWN MARGIN (SDM)
LCO 3.1.3	Moderator Temperature Coefficient (MTC)
LCO 3.1.4	Control Element Assembly (CEA) Alignment
LCO 3.1.6	Regulating Control Element Assembly (CEA) Insertion Limits
LCO 3.2.1	Linear Heat Rate (LHR)
LCO 3.2.2	Total Integrated Radial Peaking Factor (F_r^T)
LCO 3.2.4	AXIAL SHAPE INDEX (ASI)
LCO 3.4.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
LCO 3.9.1	Boron Concentration

2

INSERT 2

1. WCAP-11596-P-A, “Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” June 1988 (Westinghouse Proprietary).

2. NF-TR-95-01, “Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants,” Florida Power & Light Company, January 1995.

3. CENPD-275-P, Revision 1-P-A, “C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers,” May 1988, & Revision 1-P Supplement 1-P-A, April 1999.

4. CEN-123(F)-P, “Statistical Combination of Uncertainties Methodology Part 1: CE Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for St. Lucie Unit 1,” December 1979.

5. CEN-123(F)-P, “Statistical Combination of Uncertainties Methodology Part 3: CE Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for St. Lucie Unit 1,” February 1980.

6. CEN-191(B)-P, “CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2,” December 1981.

7. Letter, J.W. Miller (NRC) to J.R. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval of CEN-123(F)-P (three parts) and CEN-191(B)-P).

8. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), Docket No. 50-389, “St. Lucie Unit 2 – Change to Technical Specification Bases Sections ‘2.1.1 Reactor Core’ and ‘3/4.2.5 DNB Parameters’ (TAC No. M87722),” March 14, 1994 (Approval of CEN-371(F)-P).

9. CENPD-225-P-A, “Fuel and Poison Rod Bowing,” June 1983.

10. CENPD-139-P-A, “C-E Fuel Evaluation Model Topical Report,” July 1974.

11. CEN-161(B)-P-A, “Improvements to Fuel Evaluation Model,” August 1989.

12. CEN-161(B)-P, Supplement 1-PA, “Improvements to Fuel Evaluation Model,” January 1992.

13. CENPD-132, Supplement 3-P-A, “Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS,” June 1985.

14. CENPD-133, Supplement 5-A, “CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis,” June 1985.

15. CENPD-134, Supplement 2-A, “COMPERC-II, a Program for Emergency Refill-Reflood of the Core,” June 1985.

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INSERT 2

16. CENPD-135-P, Supplement 5, “STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program,” April 1977.

17. Letter, R.L. Baer (NRC) to A.E. Scherer (CE), “Evaluation of Topical Report CENPD-135, Supplement #5,” September 6, 1978.

18. CENPD-137, Supplement 1-P, “Calculative Methods for the C-E Small Break LOCA Evaluation Model,” January 1977.

19. CENPD-133, Supplement 3-P, “CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident,” January 1977.

20. Letter, K. Kneil (NRC) to A.E. Scherer (CE), “Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1 P,” September 27, 1977.

21. CENPD-138, Supplement 2-P, “PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup,” January 1977.

22. Letter, C. Aniel (NRC) to A.E. Scherer (CE), “Evaluation of Topical Report CENPD-138, Supplement 2-P,” April 10, 1978.

23. Letter, W.H. Bohlke (FPL) to Document Control Desk (NRC), “St. Lucie Unit 2, Docket No. 50-389, Proposed License Amendment, MTC Change from –27 pcm to –30 pcm,” L-91-325, December 17, 1991.

24. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), “St. Lucie Unit 2 – Issuance of Amendment Re: Moderator Temperature Coefficient (TAC No. M82517),” July 15, 1992.

25. Letter, J.W. Williams, Jr. (FPL) to D.G. Eisenhut (NRC), “St. Lucie Unit No. 2, Docket No. 50-389, Proposed License Amendment, Cycle 2 Reload,” L-84-148, June 4, 1984.

26. Letter, J.R. Miller (NRC) to J.W. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval to Methodology contained in L-84-148).

27. CEN-348(B)-P-A, Supplement 1-P-A, “Extended Statistical Combination of Uncertainties,” January 1997.

28. CEN-372-P-A, “Fuel Rod Maximum Allowable Gas Pressure,” May 1990.

29. CEN-396(L)-P, “Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/KG for St. Lucie Unit 2,” November 1989 (NRC SER dated October 18, 1991, Letter J.A. Norris (NRC) to J.H. Goldberg (FPL), TAC No. 75947).

30. CENPD-269-P, Rev. 1-P, “Extended Burnup Operation of Combustion Engineering PWR Fuel,” July 1984.

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INSERT 2

31. CEN-289(A)-P, “Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2,” December 1984 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

32. CENPD-137, Supplement 2-P-A, “Calculative Methods for the ABB CE Small Break LOCA Evaluation Model,” April 1998.

33. CENPD-140-A, “Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis,” June 1976.

34. CENPD-387-P-A, Revision 000, “ABB Critical Heat Flux Correlations for PWR Fuel,” May 2000.

35. CENPD-132, Supplement 4-P-A, “Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

36. CENPD-137, Supplement 2-P-A, “Calculative Methods for the ABB CE Small Break LOCA Evaluation Model,” April 1998.

37. CENPD-404-P-A, Rev. 0, “Implementation of ZIRLOTM Cladding Material in CE Nuclear Power Fuel Assembly Designs,” November 2001.

38. WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985.

39. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification,” February 1994.

40. WCAP-11397-P-A, (Proprietary), 'Revised Thermal Design Procedure," April 1989.

41. WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

42. WCAP-14565-P-A, Addendum 1, "Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003.

43. Letter, W. Jefferson Jr. (FPL) to Document Control Desk (USNRC), “St. Lucie Unit 2 Docket No. 50-389: Proposed License Amendment WCAP-9272 Reload Methodology and Implementing 30% Steam Generator Tube Plugging Limit,” L-2003-276, December 2003 (NRC SER dated January 31, 2005, Letter B.T. Moroney (NRC) to J.A. Stall (FPL), TAC No. MC1566).

44. WCAP-14882-P-A, Rev. 0, “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses”, April 1999.

45. WCAP-7908-A, Rev. 0, “FACTRAN-A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod”, December 1989.

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INSERT 2

46. WCAP-7979-P-A, Rev. 0, “TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code”, January 1975.

47. WCAP-7588, Rev. 1-A, “An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Special Kinetics Methods”, January 1975.

48. WCAP-16045-P-A, Revision 0, “Qualification of the Two-Dimensional Transport Code PARAGON,” August 2004.

49. EMF-96-029(P)(A), Volumes 1 and 2, “Reactor Analysis System for PWRs, Volume 1 Methodology Description, Volume 2 Benchmarking Results,” Siemens Power Corporation, January 1997.

50. XN-NF-78-44 (NP)(A), “A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors,” Exxon Nuclear Company, Inc.,” October 1983.

51. XN-75-27(A) and Supplements 1 through 5, “Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors,” Exxon Nuclear Company, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P).

52. XN-NF-82-06 (P)(A), Rev. 1 and Supplements 2, 4, and 5, “Qualification of Exxon Nuclear Fuel for Extended Burnup,” Exxon Nuclear Company, Inc., October 1986.

53. XN-NF-85-92(P)(A), “Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results,” Exxon Nuclear Company, Inc., November 1986.

54. ANF-88-133(P)(A) and Supplement 1, “Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU,” Advanced Nuclear Fuels Corporation, December 1991.

55. EMF-92-116(P)(A), Rev. 0, “Generic Mechanical Design Criteria for PWR Fuel Design,” Siemens Power Corporation, February, 1999.

56. BAW-10240(P)(A), Rev.0, “Incorporation of M5™ Properties in Framatome ANP Approved Methods,” Framatome ANP, Inc., May 2004.

57. XN-NF-82-21(P)(A), Revision 1, “Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations,” Exxon Nuclear Company, September 1983.

58. EMF-92-153(P)(A), Revision 1, “HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel,” January 2005.

59. EMF-1961(P)(A), Revision 0, “Statistical/Transient Methodology for Combustion Engineering Type Reactors,” Siemens Power Corporation, July 2000.

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60. EMF-2310(P)(A), Revision 1, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” Framatome ANP, Inc., May 2004.

61. XN-75-32(P)(A), Supplements 1, 2, 3, and 4, “Computational Procedure for Evaluating Fuel Rod Bowing,” October 1983.

62. BAW-10231P-A Revision 1, “COPERNIC Fuel Rod Design Computer Code,” January 2004.

63. EMF-2103(P)(A) Revision 0, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” April 2003.

64. EMF-2328 (P)(A) Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” March 2001.

5.6 Reporting Requirements

5.6.4 RCS Pressure and Temperature Limits Report (continued)

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

----------REVIEWER'S NOTE----------
Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the PTLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.

[Identify the NRC staff approval document by date.]

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

----------REVIEWER'S NOTE----------
The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

4

5.6 Reporting Requirements

~~5.6.4 RCS Pressure and Temperature Limits Report (continued)~~

~~7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~

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Table 3.3-6, Action 27
3.3.3.6 Actions c and d
DOC M03

5.6.~~5~~ Post Accident Monitoring Report

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When a report is required by Condition B or F of LCO 3.3.~~[11]~~, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

~~5.6.6 Tendon Surveillance Report~~

~~[Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]~~

5

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6.9.1.12

5.6.~~7~~ Steam Generator Tube Inspection Report

5

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification ~~5.5.9~~ 5.5.6, "Steam Generator (SG) Program." The report shall include:

6.9.1.12.a

a. The scope of inspections performed on each SG;

DOC M02

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

6.9.1.12.c

1. The nondestructive examination techniques utilized;

5.6 Reporting Requirements

6.9.1.12 5.6.~~7~~ 5 Steam Generator Tube Inspection Report (continued) 1

6.9.1.12.d DOC A03

2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

6.9.1.12.g DOC A03

3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment;

6.9.1.12.e

4. The number of tubes plugged ~~[or repaired]~~ during the inspection outage~~;~~. and

~~[5. The repair methods utilized and the number of tubes repaired by each repair method.]~~ 2

DOC M02

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

6.9.1.12.f 6.9.1.12.h

e. The number and percentage of tubes plugged ~~[or repaired]~~ to date, and the effective plugging percentage in each SG; 2

DOC M02

f. The results of any SG secondary side inspections~~;~~. and

~~[g. Insert any plant-specific reporting requirements, if applicable.]~~ 2

JUSTIFICATION FOR DEVIATIONS
5.6, REPORTING REQUIREMENTS

1. Changes are made (additions, deletions, and/or changes) to the ISTS that reflect the plant specific nomenclature, number, reference, system description, analysis, licensing basis, or licensing basis description.

2. The ISTS contains bracketed information and/or values that are generic to all Combustion Engineering vintage plants. The brackets are removed, and the proper plant specific information/value is provided. This is acceptable since the information/value is changed to reflect the current licensing basis.

3. The Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed into what is needed to meet this requirement. This Note is not meant to be retained in the final version of the plant specific submittal.

4. ISTS 5.6.4 provides requirements for the RCS Pressure and Temperature Limits report. PSL is not adopting a Pressure Temperature Limits Report (PTLR) and is retaining the RCS Pressure and Temperature (P/T) limits in the ITS 3.4.3, RCS Pressure and Temperature (P/T) Limits. Therefore, this report is not included in the PSL ITS, consistent with the current licensing basis. Subsequent Specifications are renumbered as a result of this deletion.

5. ISTS 5.6.6 provides requirements for the Tendon Surveillance Report. The Containment design at PSL does not include pre-stressed concrete tendons. Therefore, this report is not included in the PSL ITS, consistent with the current licensing basis. Subsequent Specifications are renumbered as a result of this deletion.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.6, REPORTING REQUIREMENTS

There are no specific No Significant Hazards Considerations for this Specification.

ATTACHMENT 7

ITS Section 5.7, High Radiation Area

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

ADMINISTRATIVE CONTROLS

6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls

and b

5.7 6.12.1 ~~In lieu of the "control device" or "alarm signal"~~ required by paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr measured at a distance of 30 cm (12 in) shall be barricaded and conspicuously 5.7.1 posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An ~~health physics qualified~~ individual ~~(i.e.,~~ qualified in radiation protection procedures~~)~~ with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency ~~specified by the facility Health Physicist~~ in the RWP.

A02

LA01

LA02 A02

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rad/hr at 1 meter shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the ~~Shift Foreman on duty and/or health physics supervision~~. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in 5.7.2 the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

shift supervisor, radiation protection manager, or his or her designee.

radiation protection

radiation protection

A02

5.7.1 * ~~Health Physics~~ personnel or personnel escorted by ~~Health Physics~~ personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

ADMINISTRATIVE CONTROLS

6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls

5.7 6.12.1 ~~In lieu of the "control device" or "alarm signal"~~ required by paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr measured at a distance of 30 cm (12 in) shall be barricaded and conspicuously 5.7.1 posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have bee made knowledgeable of them.

c. An ~~health physics qualified~~ individual ~~(i.e.,~~ qualified in radiation protection procedures~~)~~ with a radiation dose rate monitoring device who is responsible for providing position control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified ~~by the facility Health Physicist~~ in the RWP.

A02 LA01

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rad/hr at 1 meter shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the ~~Shift Foreman on duty and/or health physics supervision~~. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in 5.7.2 that area. For individual areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

LA02 A02

shift supervisor, radiation protection manager, or his or her designee.

radiation protection

radiation protection

A02

* ~~Health Physics~~ personnel or personnel escorted by ~~Health Physics~~ personnel 5.7.1 shall be exempt for the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

DISCUSSION OF CHANGES
ITS 5.7, HIGH RADIATION AREA

ADMINISTRATIVE CHANGES

A01 In the conversion of the St. Lucie Plant (PSL) Unit 1 and Unit 2 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG - 1432, Rev. 5.0, "Standard Technical Specifications - Combustion Engineering Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A02 CTS 6.12 specifies “health physics” personnel when referring to radiation protection qualified individuals, supervision, and personnel exempt from radiation work permits (RWPs). PSL does not use the term “health physics” when referring to these personnel but rather uses the term “radiation protection.” ITS 5.7 replaces the term “health physics” with “radiation protection.” This changes the CTS by replacing references to health physics personnel with radiation protection personnel. Specifically, a health physics qualified individual is replaced with an individual qualified in radiation protection procedures; health physics supervision is replaced with radiation protection manager, or his or her designee; and health physics personnel is replaced with radiation protection personnel. These changes are consistent with the ISTS and are designated as administrative because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 (*Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements*) CTS 6.12.1.c, states, in part, that the health physics qualified individual responsible for positive control over the activities within the high radiation area shall also perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the RWP. ITS 5.7.1.c provides a similar requirement but does not specify that the frequency of radiation monitoring specified in the RWP is specified by the facility Health Physicist. This changes the CTS by moving the procedural detail that the frequency of radiation monitoring is specified by the facility Health Physicist to the Updated Final Safety Analysis Report (UFSAR).

The removal of these details, which are related to who establishes the frequency of radiation monitoring in the RWP, from the Technical Specifications is

acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. Implementation of radiation work procedures, including RWPs, are the responsibility of the radiation protection manager (i.e., facility Health Physicist). This change is acceptable because the individual qualified in radiation protection procedures (i.e., health physics qualified individual) who is responsible for providing positive control over the activities within the area will continue to be required to perform periodic radiation surveillance at the frequency specified in the RWP independent of who is responsible for establishing the surveillance frequency. This change is also acceptable because the removed information will be adequately controlled in the UFSAR. Any changes to the UFSAR are made pursuant to 10 CFR 50.59 or 10 CFR 50.71(e). These change control processes ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

LA02 (*Type 4 – Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAP, CLRT Program, IST Program, ISI Program or Surveillance Frequency Control Program*) CTS 6.12.2 uses the title "Shift Foreman." ITS 5.7.2 uses the generic title "shift supervisor." This changes the CTS by moving the specific PSL organizational title to the UFSAR or the Quality Assurance Topical Report (QATR) and replacing it with a generic title.

The removal of these details, which are related to meeting Technical Specification requirements, from the Technical Specifications is acceptable because this type of information is not necessary to be included in Technical Specifications to provide adequate protection of public health and safety. The allowance to relocate the specific PSL organizational titles out of the Technical Specifications is consistent with the NRC letter from C. Grimes to the Owners Groups Technical Specification Committee Chairmen, dated November 10, 1994. The various requirements of the shift supervisor are retained in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the UFSAR or the QATR. Any changes to the UFSAR are made pursuant to 10 CFR 50.59 or 10 CFR 50.71(e) and changes to the QATR are made pursuant to 10 CFR 50.54(a). These change control processes ensure changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information related to meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

6.12.1 As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

Insert 1

1

~~5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation~~

~~a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.~~

~~b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification or radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measurers.~~

~~c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.~~

~~d. Each individual or group entering such an area shall possess:~~

~~1. A radiation monitoring device that continuously displays radiation dose rates in the area, or~~

~~2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or~~

~~3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or~~

~~4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,~~

1

Insert 1

6.12.1 including footnote*

5.7.1 Each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr measured at a distance of 30 cm (12 in) shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt for the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the RWP.

6.12.2

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rad/hr at 1 meter shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued) 1

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued) 1

c. Individuals qualified in radiation protection procedures my be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued) 1

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

6.12.1 As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

Insert 1

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~~5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation~~

~~a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.~~

~~b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification or radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measurers.~~

~~c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.~~

~~d. Each individual or group entering such an area shall possess:~~

~~1. A radiation monitoring device that continuously displays radiation dose rates in the area, or~~

~~2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or~~

~~3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or~~

~~4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,~~

1

Insert 1

6.12.1 including footnote*

5.7.1 Each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr measured at a distance of 30 cm (12 in) shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt for the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the RWP.

6.12.2

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rad/hr at 1 meter shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued) 1

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued) 1

c. Individuals qualified in radiation protection procedures my be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued) 1

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

JUSTIFICATION FOR DEVIATIONS
ITS 5.7, HIGH RADIATION AREA

1. ISTS 5.7 provides requirements for High Radiation Areas. The brackets are removed and the proper plant specific information/value is provided. ITS 5.5.7 is revised to reflect the St. Lucie Plant Unit 1 and Unit 2 current licensing basis and high radiation area controls The change is consistent with the requirements in CTS 6.12.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 5.7, HIGH RADIATION AREA

There are no specific No Significant Hazards Considerations for this Section.

ATTACHMENT 8

Relocated/Deleted Current Technical Specifications (CTS)

CTS 6.11, Radiation Protection Program

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

~~6.11 RADIATION PROTECTION PROGRAM~~

LA01

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

~~6.11 RADIATION PROTECTION PROGRAM~~

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

LA01

DISCUSSION OF CHANGES
CTS 6.11, RADATION PROTECTION PROGRAM

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 6 - Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS 6.11 provides requirements for the Radiation Protection Program. The ITS does not include these requirements. This changes the CTS by moving the requirements for the Radiation Protection Program to the UFSAR.

The removal of these requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The Radiation Protection Program requires procedures to be prepared for personnel radiation protection consistent with 10 CFR 20. These procedures are for nuclear plant personnel and have no impact on nuclear safety or the health and safety of the public. Requirements to have procedures to implement 10 CFR 20 are contained in 10 CFR 20.1101(b). Periodic review of these procedures is addressed in 10 CFR 20.1101(c). Since the PSL Units 1 and 2 Operating Licenses require compliance with 10 CFR 20, there is no need to repeat the requirements in the ITS. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 6.11, RADIATION PROTECTION PROGRAM

There are no specific No Significant Hazards Considerations for this Specification.

CTS 6.13, Process Control Program

Current Technical Specifications (CTS) Markup and Discussion of Changes (DOCs)

~~ADMINISTRATIVE CONTROLS~~

~~6.13 PROCESS CONTROL PROGRAM (PCP)~~

~~Changes to the PCP:~~

~~1. Shall be documented and this documentation shall contain:~~

~~a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and~~

~~b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.~~

~~2. Shall become effective after the approval of the plant manager.~~

LA01

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

1. Shall be documented and this documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after the approval of the plant manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

See ITS 5.5

~~ADMINISTRATIVE CONTROLS~~

~~6.13 PROCESS CONTROL PROGRAM (PCP)~~

~~Changes to the PCP:~~

~~1. Shall be documented and this documentation shall contain:~~

LA01

~~a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and~~

~~b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.~~

~~2. Shall become effective after the approval of the plant manager.~~

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

1. Shall be documented and this documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after the approval of the plant manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

See ITS 5.5

DISCUSSION OF CHANGES
CTS 6.13, PROCESS CONTROL PROGRAM (PCP)

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA01 *(Type 6 - Removal of LCO, SR, or other TS requirements to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS Definition 1.23 contains the definition for the Process Control Program (PCP). CTS 6.13 describes the process for control of changes to the PCP. The ITS does not include these requirements. This changes the CTS by moving the requirements of the PCP to the UFSAR.

The removal of these requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71. Compliance with these regulations is required by the PSL Units 1 and 2 Operating Licenses, and procedures are the method to ensure compliance with the program. Regulations provide an adequate level of control for the affected requirements and inclusion of this requirement in the Technical Specifications is not necessary. Also, this change is acceptable because these details will be adequately controlled in the UFSAR. Any changes to the UFSAR are made under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of requirements because details for meeting Technical Specification and regulatory requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 6.13, PROCESS CONTROL PROGRAM (PCP)

There are no specific No Significant Hazards Considerations for this Specification.

ENCLOSURE 3

LICENSEE IDENTIFIED CHANGES THAT MAY REQUIRE TECHNICAL BRANCH REVIEW

(2 TOTAL PAGES, INCLUDING COVER SHEETS)

LICENSEE IDENTIFIED CHANGES THAT MAY REQUIRE TECHNICAL BRANCH REVIEW

Changes included in the Improved Technical Specifications (ITS) conversion submittal that are not consistent with the Current Technical Specifications (CTS) and are not the result of adopting the Improved Standard Technical Specifications (ISTS) as described in NUREG-1432, Rev. 5.0. The following is a list of changes identified in Enclosure 2 that meet this criterion in the St. Lucie Plant (PSL), Unit 1 and Unit 2 ITS conversion submittal, but do not involve a design change to the plant:

1. Change to Engineered Safety Features Actuation System (ESFAS) Main Steam Isolation Signal (MSIS) manual and automatic actuation logic Applicability and action change to match the supported system's Technical Specifications actions. The change is not technical in nature, but rather aligns the MSIS instrumentation actions with the supported system actions. (Refer to Enclosure 2, Volume 8, ITS Section 3.3.4, Discussion of Change (DOC) L01)

2. Add risk informed completion time (RICT) to Unit 1 ESFAS automatic actuation logic and Unit 2 ESFAS MSIS manual actuation. The change is supported by the existing RICT Program and configuration risk management program previously approved in License Amendments 247 and 199, for Unit 1 and Unit 2, respectively. Unit 2 ESFAS automatic actuation logic already applies RICT and both Unit 1 and Unit 2 already apply RICT to manual actuation channels. (Refer to Enclosure 2, Volume 8, ITS 3.3.4 DOCs L04 and L05)

3. Add previously approved program requirements in the Unit 2 Reactor Coolant Pump (RCP) Flywheel Program in addition to the current requirements based on the RCP Flywheel material. In addition, RCP Flywheel Program are proposed for the Unit 1 Technical Specifications and include the same requirements based on RCP flywheel material. Both testing methods have been previously approved by the NRC for PSL. (Refer to Enclosure 2, Volume 16, ITS 5.5, DOCs M01 and L02)

4. The periodic frequency of selected Surveillances in ITS Section 5.5 are proposed to be relocated to the Surveillance Frequency Control Program (SFCP). This deviation from ISTS Rev. 5 and Technical Specification Task Force (TSTF) Traveler TSTF-425 has been previously approved for at least three (3) plants. PSL currently has an approved SFCP. (Refer to Enclosure 2, Volume 16, ITS 5.5, DOC LA01)

ENCLOSURE 4

DISPOSITION OF EXISTING LICENSE AMENDMENT REQUESTS

(2 TOTAL PAGES, INCLUDING COVER SHEETS)

DISPOSITION OF EXISTING LICENSE AMENDMENT REQUESTS

The following License Amendment Request was recently approved the NRC. The following table describes the request, and its effect on the ITS conversion, and its disposition. R1

DISPOSITION OF EXISTING LICENSE AMENDMENT REQUESTS				
Submittal Date	Description of Change	Affected ITS Submittal Specifications	Affected CTS Pages	Disposition
December 21, 2020	Permit the application of risk-informed completion times (RICT) for the 120-Volt AC Instrument Bus requirements, consistent with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times RITSTF Initiative 4b".	ITS 3.8.7 ITS 3.8.9	Unit 1 pg. 3/4 8-8 Unit 2 pg. 3/4 8-15	License Amendments 252 (Unit 1) and 207 (Unit 2) issued on January 14, 2022. R1

ENCLOSURE 5

REGULATORY COMMITMENTS

(2 TOTAL PAGES, INCLUDING COVER SHEETS)

REGULATORY COMMITMENTS

No.	Commitments for TSTF-500	Due Date/Event
1	Revise or develop plant procedures, as applicable, to address the following: • minimum required procedural time to measure battery float current is at least 30 seconds or as recommended by the float current measurement instrument manufacturer. This minimum float current measurement time is required to provide a more accurate battery float current reading. • routinely monitor battery room temperature such that a room temperature excursion could reasonably expect to be detected and corrected prior to the average battery electrolyte temperature dropping below the minimum electrolyte temperature. • verification of the selection of the pilot cell or cells when performing the Surveillance that verifies the float voltage of each connected battery cell. • ensure that the modified performance discharge test completely encompasses the load profile of the battery service test and that it adequately confirms the intent of the service test to verify the battery capacity to supply the design basis load profile.	Upon Implementation
2	Relocate the following requirements to the Battery Monitoring and Maintenance Program and implementing procedures: • Battery cell resistance limits in existing Surveillance • Monitoring of battery parameters (i.e., specific gravity, electrolyte level, cell temperature, float voltage, connection resistance, and physical condition)	Upon Implementation

ENCLOSURE 6

LIST OF REQUIRED UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) DESCRIPTIONS FOR TSTF 500

(2 TOTAL PAGES, INCLUDING COVER SHEETS)

LIST OF REQUIRED UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) DESCRIPTIONS FOR TSTF-500

The following table identifies UFSAR descriptions for the 125 VDC batteries required by St. Lucie Plant (PSL), Units 1 and Unit 2, as part of the adoption of TSTF-500, "DC Electrical Rewrite - Update to TSTF-360," Revision 2. These changes will be included with the required implementation date in the Issuance of Amendment letter.

REQUIRED UFSAR DESCRIPTION	DUE DATE/EVENT
PSL will change or verify that the UFSAR: 1. Describes how a 5 percent design margin for the batteries corresponds to a 2 amp float current value indicating that the battery is 95 percent charged. 2. States that long term battery performance is supported by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer. 3. Describes how the batteries are sized with correction margins that include temperature and aging and how these margins are maintained. 4. States the minimum established design limit for battery terminal float voltage. 5. States the minimum established design limit for electrolyte level. 6. States the minimum established design limit for electrolyte temperature. 7. Describes how each battery is designed with additional capacity above that required by the design duty cycles to allow for temperature variations and other factors. 8. Describes normal DC system operation, i.e., powered from the battery chargers with the batteries floating on the system, and a loss of normal power to the battery charger describing how the DC load is automatically powered from the station batteries.	Upon implementation (applies to all)